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TECHNICAL PROGRESS REVIEWS

POWER REACTOR TECHNOLOGY

Prepared for U. S. ATOMIC ENERGY COMMISSION by GENERAL NUCLEAR ENGINEERING CORP.

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TECHNICAL PROGRESS REVIEWS

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POWER REACTOR TECHNOLOGY

a review of recent developments *prepared by*

GENERAL NUCLEAR ENGINEERING CORP.

SEPTEMBER 1958 VOLUME 1 NUMBER 4

foreword

This quarterly review of reactor development has been prepared at the request of the Division of Information Services of the U. S. Atomic Energy Commission. Its purpose is to assist interested organizations in the task of keeping abreast of new results in reactor technology for civilian application.

The Review is a concise discussion of selected phases of research and development for which there have been significant advances or heightened interest in the past few months. It is not meant to be a comprehensive abstract of all material published during the quarter nor is it meant to be a treatise on any part of the subject. The intention is to cover the various areas of reactor development from the general viewpoint of the reactor designer rather than from the more detailed points of view of specialists in the individual areas. However, papers that are thought to be of particular significance or of particular usefulness in specialized fields will be mentioned in short notes. In the over-all plan of the report, it is intended that various subjects will be treated and brought up to date from time to time.

Any interpretation of results which is given represents only the opinion of the editors of the report, who are General Nuclear Engineering Corporation personnel. Readers are urged to consult the original references in order to obtain all the background of the work reported and the interpretation of the results given by the original authors.

W. H. ZINN
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GENERAL RESEARCH AND DEVELOPMENT

REACTOR PHYSICS

Lattice Measurements and Analyses

Slightly Enriched Uranium Metal and UO_2 Light-water-moderated Lattices

During the past several years an extensive program has been undertaken at Bettis to measure the properties of slightly enriched uranium metal and UO_2 light-water-moderated lattices. The primary objective of the program has been to furnish experimental information for the analysis of the Shippingport Pressurized Water Reactor (PWR).

Measurements involving the uranium metal fuel have been carried out jointly with BNL.[1] Measurements of the thermal utilization, resonance escape probability, and fast-fission factor for 1.30 per cent enriched uranium metallic fuel for rods 0.600 in. in diameter have been previously reported by Kranz.[2]

Measurements of the lattice parameters for 1.15 and 1.3 per cent enriched uranium metal and 1.3 per cent enriched uranium oxide (density, 7.5 and 10.53 g/cm^3) rods of 0.600- and 0.387-in. diameters have recently been reported by Klein et al.[3] Details of the measurements and the results are given in this reference. The following conclusions are drawn from the results:

1. The measured fast effect appears to be independent of rod size and fuel composition and is solely dependent on the water-to-uranium ratio. In close-packed lattices there is a very strong coupling between fuel rods, and, in fact, only 10 per cent of the U^{238} fissions in a rod are caused by neutrons born in the rod. The strong coupling causes the fast-fission factor to behave as if the lattice were homogeneous.

2. Thermal utilization calculations yielded the well-known result that diffusion theory gives too high a value (about 2 per cent in these cases) if the diffusion theory is applied to both fuel and moderator. It is indicated that a somewhat better answer is achieved if diffusion theory is used only in the moderator and transport theory is used in the fuel region to determine the boundary conditions at the fuel-moderator interface (method of successive generations).[4]

3. Results of the resonance escape measurements were in rather good agreement with a Monte Carlo method developed at Bettis. In this method the 18 resolved resonances of U^{238} are treated individually, with a constant cross section of 5.5 barns being added to account for the unresolved resonances and 1/v absorption. The simple resonance escape probability prescription $p = e^{-1/T}$, where 1/T is proportional to the resonance integral and the depression of resonance flux near the fuel is neglected, gave values of p which were higher than those measured. The effective resonance integrals used for these calculations were (for uranium metal)

$$RI = 9.25\left(1 + 2.67\,\frac{S}{M}\right)$$

and (for UO_2)

$$RI = 11.5\left(1 + 1.92\,\frac{S}{M}\right)$$

These resonance integrals include the contribution of 1/v absorption. The Dancoff correction[5] for close-packed lattices, ranging in magnitude for these lattices from 5 to 25 per cent, was applied. The results show that better agreement with experiments using the simple theory would be obtained if the Dancoff correction were omitted.

Kinetic and buckling measurements on the identical lattices described have been performed

by Brown et al.[6] For these lattices measurements have been made of critical buckling, radial and axial reflector savings, and the temperature coefficient of reactivity in the range 20 to 50°C. Migration area and migration-area anisotropy were also deduced from changes of buckling with reactivity.

The results of the experiments indicate that the reflector saving for a water reactor is a slowly varying function of the core composition or core leakage and is independent of the core size for reasonably large cores. For metallic uranium–water lattices in which the neutron age is quite insensitive to uranium-water composition, the experiments indicate a value of approximately 7.1 cm for the reflector saving at room temperature. For UO_2 lattices, the saving varies between 7.3 and 8.5 cm at room temperature, depending on the water-to-fuel ratio and the UO_2 density. The temperature-coefficient measurements showed a positive temperature coefficient at 20°C for the uranium-metal lattice having a water-to-fuel ratio of 3, although a negative value was measured at 40°C. All other lattices measured were negative in the 20 to 40°C temperature interval. The anisotropy of the migration area, the ratio of migration area in the direction parallel to the rod to that in the direction perpendicular to the rod, was very nearly unity.

D_2O-moderated Lattices

An extensive set of buckling measurements on lattices in D_2O, made up of uranium metal plates, uranium metal rods, and UO_2 rods, all of natural enrichment, has been made at Savannah River. These measurements and their results are summarized in references 7 and 8. (See also references 9 and 10.) The data, along with the Canadian measurements,[11] the measurements which have been made at Hanford in connection with the plutonium recycle program,[12] the Savannah River measurements summarized in reference 13, and the earlier measurements on single rod lattices[14] make up a very extensive body of information on D_2O-moderated lattices. The data are too voluminous to be summarized here, but some interesting aspects of the measurements are mentioned.

The Savannah River measurements show that the curves of buckling as a function of moderator-to-fuel ratio are rather similar for metallic and oxide fuel, provided the plot is made on the basis of the ratio of fuel volume (either metal or oxide) to moderator volume. To make a quantitative comparison, bucklings for a set of lattices composed of bare UO_2 plates in D_2O were calculated, using methods tested against the measured bucklings for other UO_2 lattices, and compared with the bucklings of similar bare uranium metal lattices.[8] It was found that in the range of volume ratios around 20 to 25, the bucklings for the oxide lattices differed very little from those of the metal lattices. At higher moderator-to-fuel ratios the uranium metal lattices gave higher bucklings, whereas at lower moderator-to-fuel ratios the oxide lattices gave higher bucklings. The moderator-to-fuel ratio in the range 20 to 25 is not far from the ratio giving maximum buckling. Consequently, it would appear feasible to design a reactor in which metal and oxide fuel elements would be interchangeable, at least from the standpoint of reactor physics.

In reference 7 the measured bucklings for lattices of metal plates were used with computed constants in the criticality equation to determine a value for the resonance escape probability in the various lattices. From the resonance escape probability, an expression was determined for the effective resonance integral of uranium metal as a function of the effective surface to mass ratio (S/M).

$$RI = 8.5 + 24\ S/M$$

Graphite-moderated Lattices

Measurements of the lattice parameters for the natural-uranium graphite-moderated water-cooled lattices have been reported by Nilson.[15] Measurements were made on solid and cored slugs of 1.33, 1.68, and 1.92 in. outer diameter, with and without coolant. Some measurements with an organic coolant, monoisopropylbiphenyl, $C_{15}H_{16}$, are also included.

The results of the thermal utilization measurements are in good agreement with calculations by the P_3 approximation and by the method of successive generations.[4] The fast effect measurements agree both with previous measurements and with calculations based on a suitably averaged one-group expression. The theoretical values of resonance escape based on the simple prescription $p = e^{-1/T}$ support the results of Hellstrand.[16]

An analysis has been reported in reference 17 of the earlier exponential measurements on graphite lattices by Davenport.[18]

If the criticality equation is written as

$$\eta \epsilon pf = (1 + L^2B^2)\, e^{\tau B^2}$$

and, if the resonance escape probability p is given by the expression

$$p = \exp -\left(\frac{N_f V_f \bar{\phi}_f}{N_m V_m \xi_m \sigma_m \bar{\phi}_m} RI_{eff}\right)$$

then the criticality relation may be written as

$$\ln\left(\frac{1 + L^2B^2}{f}\right) + \tau B^2 = -\frac{N_f V_f \bar{\phi}_f}{N_m V_m \xi_m \sigma_m \bar{\phi}_m} RI_{eff} + \ln \eta\epsilon$$

where N is the atom density, V is the volume of material, $\bar{\phi}$ is the average resonance flux, ξ is the average logarithmic energy loss per collision, σ is the scattering cross section, and the subscripts f and m are fuel and moderator, respectively. If bucklings are measured for several values of V_f/V_m for a given fuel-rod size and shape, the left-hand side of the equation, incorporating the measured values of B^2, may be plotted as a function of the quantity

$$\frac{N_f V_f \bar{\phi}_f}{N_m V_m \xi_m \sigma_m \bar{\phi}_m}$$

The effective resonance integral is then given by the slope of the line, and the value of $\eta\epsilon$ can be determined from the intercept. When this analysis was applied to the experimental data, using calculated values of f, L^2, and τ, values of η were found which varied slightly with fuel rod size, from 1.3163 (for a rod of 0.925 in. diameter) to 1.3083 (for rods of 1.66 in. diameter). The effective resonance integral for rods of various surface-to-mass ratio was found to be represented by the expression

$$RI_{eff} = 7.10\ (1 + 3.90\ S/M)$$

This is the resonance integral with the 1/v absorption omitted. This expression gives resonance integral values that are in the range of those currently in use. Although the expression is different in form from that of Hellstrand,[16] it gives values not greatly different over the 0.05 to 0.10 surface-to-mass range.

Highly Enriched Lattices

The results of zero-power experiments on the Army Package Power Reactor (APPR) are recorded in references 19 and 20. Criticality measurements, control-rod worth measurements, temperature coefficients, and flux distributions are described in considerable detail. This reactor is fueled with highly enriched uranium and has a rather high macroscopic absorption cross section since the structural material is stainless steel. The reactor is moderated and reflected by ordinary water. Although this reactor type is one that has limited applications, the data are useful in a general way and for certain problems which arise occasionally and have not previously had extensive experimental investigation. In particular, surveys of neutron flux and gamma intensity through the thermal shield are given in some detail. These quantities are quite difficult to calculate from first principles, and the amount of data that have previously been available for checking calculation methods have been extremely limited. Other measurements that may be of interest include flux distributions around control rods, temperature coefficients, and flux distributions in the core when the water level is dropped sufficiently to expose part of the core.

Reference 21 contains a description of analyses of water-moderated highly enriched critical experiments (pool type reactor) by two-group and thirty-group calculations. They indicate the areas of uncertainty which exist in the theory of this relatively simple reactor type.

Effective Resonance Integral in Uranium Metal and Oxide

A series of measurements of the effective resonance integral in uranium metal and UO_2 has been reported by Hellstrand.[16] The measurements were performed on simple rods, on a few tubes, both empty and filled with moderator, and on rod clusters. Since the geometrical configurations measured are of interest in applied reactor design, these data should prove

to be extremely useful. The results of the measurements for rods of uranium metal and oxide show that the resonance integral may be written as a linear function of $(S/M)^{1/2}$, where S is the surface area of the rod and M is the mass of the rod over a wide range of surface to mass. The effective resonance integrals are (for uranium metal)

$$RI_{eff} = 2.81 + 24.7\left(\frac{S}{M}\right)^{1/2} \qquad 0.07 \leq \frac{S}{M} \leq 0.53$$

and (for UO_2)

$$RI_{eff} = 4.15 + 26.6\left(\frac{S}{M}\right)^{1/2} \qquad 0.08 \leq \frac{S}{M} \leq 0.7$$

These resonance integrals have been reduced from the measured values to eliminate the portion of the integral which would result from 1/v absorption. The correction applied amounts to 1.1 barns.

The concept of breaking the resonance absorption down into a volume term and a surface term was also investigated. The data, which then must be fit by a linear function of S/M, require the application of corrections for both the volume and the surface absorption terms; this formulation is found to have a narrower range of validity than the $(S/M)^{1/2}$ formulation.

For comparison with the experimental results, calculations of the resonance integral were made for the tubes and clusters, both with and without internal moderator. For these calculations a surface term is assumed which depends on the sum of the outer surface plus a fraction of the inner surface. For clusters the outer surface is assumed to be equal to the surface defined by a rubber band encircling the cluster. The amount of inner surface is calculated by a method derived by Carlvik and Pershagen.[22] There is good agreement with the experimental results. However, the measurements on fuel clusters have been made with a cadmium sleeve around the entire cluster rather than around each rod. This means that, for the case in which moderator is contained within the cluster, epi-cadmium neutrons can enter the cluster and be slowed down by the moderator to make sub-cadmium captures in the rod. No correction has been made for this effect.

Effective Resonance Integrals for Thorium and Thorium Oxide

Reference 23 reports measurements of the effective resonance integrals for thorium and thorium oxide by the danger coefficient method. The measurements were made by inserting the sample in a cadmium tube in a swimming pool type reactor. The resonance integrals determined have been related to the surface-to-mass ratio by expressions of two different types. The results are as follows.

For thorium: $0.1 \leq S/M_{Th} \leq 1.3$

$$RI_{eff} = 8.4 + 24.9\ S/M_{Th}$$

or

$$RI_{eff} = 2.9 + 24.1\ \sqrt{S/M_{Th}}$$

For ThO_2: $0.2 \leq S/M_{Th} \leq 1.3$

$$RI_{eff} = 10.4 + 27.4\ S/M_{Th}$$

or

$$RI_{eff} = 0.1 + 35.2\ \sqrt{S/M_{Th}}$$

The effective resonance integrals include the contribution of the 1/v absorption. The 1/v contribution is calculated to be 3.5 barns on the basis of a thorium thermal cross section of 7.4 barns. M_{Th} refers to the mass of thorium in the sample.

Effective Cross Sections of Higher Plutonium Isotopes

In any calculation of the long-term reactivity changes in a reactor, the isotope Pu^{240} is both an important consideration and a troublesome one. The difficulty results from the presence of a large, sharp absorption resonance at 1 ev. Consequently, the average cross section is quite sensitive to the neutron-energy spectrum, particularly in the low epithermal range. Furthermore, the absorption may be strongly self-shielded as soon as the concentration of the isotope becomes appreciable. These difficulties are discussed in reference 24, and methods of handling them are discussed. It may become a complicated matter to treat the variations in effective cross section in a reactor as burn-up proceeds since both the neutron spectrum and the self-shielding of the Pu^{240} will change with

time. The reference discusses, in particular, methods for arriving at a constant effective cross section that may be used with reasonable accuracy for many reactivity lifetime calculations. It is pointed out that for a given concentration of Pu^{240} in a given fuel element, the effective cross section of Pu^{240} can be well represented as a function of the ratio S/Z, where S is the effective moderating power per fuel atom and Z is the number of fast neutrons reaching the resonance energy region per unit volume per unit of thermal neutron flux.

$$S = \frac{\xi \Sigma_s \left(\frac{V_m}{V_f}\right) F}{N_f}$$

where F is the resonance flux disadvantage factor

$$Z = \frac{\epsilon \Sigma_i \eta_i \Sigma_{ai}}{1 + \tau B^2}$$

The effective cross section varies from about 2500 barns for S/Z = 2 to about 700 barns for S/Z = 10. Although, for a given S/Z the effective cross section will become smaller with increasing fuel irradiation, it was found that for most practical purposes a single average curve for the cross section as a function of S/Z could be used, which would give a value correct to within about ±100 barns.

Reference 25 contains the results of recent measurements of the Pu^{240} total cross section in the region of the resonance and of measurements of the Pu^{241} fission cross section over the range 0.025 to 0.50 ev.

Fission-product Decay Energy

The radiation energy emitted by fission products is of importance in determining the shutdown cooling requirements of reactor fuels and shielding requirements of reactors after shutdown and of fuels during their decay periods and during handling. In reference 26, the total disintegration rates, the rates of beta- and gamma-energy release, and the gamma-ray energy spectrum have been calculated for fission products due to thermal neutron fission of U^{235}. Reactor operating times of 1, 10, 100, and 1000 hr are treated, and the results are plotted for decay times ranging from 10^2 to 10^8 sec. The calculations are made by summing the appropriate quantities for the individual fission products. The gamma-ray spectrum is divided into seven energy groups to cover the range from 0.1 Mev to the highest energy group, which includes all gammas of energy greater than 2.6 Mev. The results are compared with previous calculations and with experimental measurements of total energy release, and they are found to be in satisfactory agreement.

Reference 27 presents computational results that are useful in estimating the fission-product activity resulting from the cyclic operation of a reactor. Such estimates are often necessary; e.g., in connection with research reactors which are operated on an 8-hr-day basis or on a 24-hr-day basis with week-end shutdown.

Briefly Noted Reports

Reference 28 is a brief discussion of critical mass calculations of the type involved in determining safe configurations for the storage, transportation, and handling of fissionable materials.

Reference 29 reports measurements of the neutron spectrum from a commercial mock-fission source. The spectrum is compared to the spectrum of primary fission neutrons and to various spectra from reactors and experimental critical assemblies.

References 30 and 31 give recent data on nuclear characteristics and reactions. Reference 30 is a compilation by the Nuclear Data Group of the National Research Council, and reference 31 is a group of papers presented at the International Conference on Neutron Interactions with the Nucleus, at Columbia University, September 1957.

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Transition Boiling and Burnout

The maximum heat flux that can be supported between a solid surface and water by a reasonably low-temperature difference is a quantity of great significance for water-cooled reactors and many other applications of heat transfer to water. In some water-cooled reactor designs this maximum heat flux sets the design limit for the power density of the reactor. Since the oxide fuel elements, which have rather low thermal conductivity, have been widely adopted, the heat-flux limitation has become somewhat less important as a determining factor on power density, but it is still the limitation for those water-cooled reactors that do not use oxide fuels and must be considered in the oxide-fueled cases, both as a possible limitation on the normal design power and as a factor that becomes important if the normal design power is inadvertently exceeded.

It is well known that several heat-transfer regimes, characterized by different relations between heat flux and driving temperature, exist in the solid-liquid heat-transfer system. The quantitative characteristics of these regimes may vary widely from reactor to reactor, depending upon the flow characteristics, the configuration of heat-transfer surfaces, and whether or not the reactor is designed to operate normally as a boiling reactor or as a subcooled pressurized reactor. However, the qualitative characteristics of the different regimes are recognizable in all cases. If it is assumed that one begins with a situation in which the liquid and the solid surface are both at a temperature below the liquid saturation temperature and in which the heat flux is relatively low, the various regimes that may be recognized as the heat flux is gradually increased may be described as follows:

1. So long as the solid surface temperature is below the fluid saturation temperature, the heat transfer is by convection, either forced or natural, as provided in the system. The heat-transfer relations in this region are well known and obey well-known correlation rules. The heat flux increases as the temperature difference between the solid surface and the liquid is increased.

2. As the surface temperature and heat flux are increased to the point where the surface temperature exceeds the saturation temperature by a few degrees, a thin layer of superheated liquid is formed adjacent to the surface, and bubbles of steam nucleate and grow in this layer from preferred spots on the solid surface. This regime is referred to as "nucleate boiling." Because of the agitation produced by the bubbles in the superheated layer near the surface, the thermal resistance of the layer is greatly reduced. As the surface temperature is increased, the bubble population in the layer increases sharply, and the heat flux rises very rapidly with increasing surface temperature; i.e., the heat-transfer coefficient for nucleate boiling is very large. As the surface temperature is increased further, the bubbles become so numerous that their motions interact. Under these conditions the nucleate-boiling heat flux reaches its peak, and, if the temperature is increased further, a new regime, "transition boiling," is entered. The maximum heat flux reached in the nucleate-boiling range is, for practical reactor cases, the maximum attainable without overheating the fuel elements. The temperature difference between the solid surface and the liquid is characteristically relatively low — of the order 100°F. Attempts to increase the heat flux above the maximum of the nucleate-boiling range will result in very much higher surface temperatures as indicated below.

3. Whereas the boiling in the nucleate range was characterized by the contact of liquid with the surface and by the formation of steam bubbles at preferred points on the surface, the transition boiling exhibits quite different characteristics. Experimental observations by Westwater and Santangelo[1] are quoted in reference 2 as follows:

> ... This boiling is entirely different from both nucleate boiling and film boiling. No active nuclei exist. In fact no liquid-solid contact exists

either. The tube is completely blanketed by a film of vapor, but the film is not smooth nor stable. The film is irregular and is in violent motion. Vapor is formed by sudden bursts at random locations along the film. Liquid rushes in toward the hot tube, but before the two can touch, a miniature explosion of vapor occurs and the liquid is thrust back violently. The newly-formed slug of vapor finally ruptures, and the surrounding liquid surges toward the tube. This process is repeated indefinitely.

The frequency of the vapor bursts is surprisingly high. For an over-all temperature difference between surface and liquid of 133°F (and a heat transfer coefficient of 164 Btu/hr ft^2 °F) each inch of the photographed side of the tube exhibited 84 bursts per second. The bursts occur so suddenly and unexpectedly that even in slow motion they resemble explosions.

Quantitatively, the unique characteristic of the transition-boiling range is that the heat flux becomes lower as the surface temperature is increased.

4. As the surface temperature is increased still more, a minimum value of heat flux is finally reached at which film boiling starts. The film-boiling regime is characterized by an orderly discharge of large bubbles from the surface with a regular frequency and at regular intervals. In this region the heat flux increases with an increase of surface temperature but at a rate much slower than that in the nucleate-boiling range. In this range the heat-transfer surface is blanketed by a relatively stable layer of steam, across which heat is transferred by conduction and radiation.

Since the maximum heat flux of the nucleate-boiling region represents the limit of heat transfer for practical cases, studies have been made in the past of the heat transfer in this regime. Fundamental studies have been handicapped by the circumstance that the surface conditions affect the characteristics quite markedly. Engineering type tests of burnout heat flux have been made which give practical power limitations. Correlations of such tests have suffered from lack of understanding of the basic heat-transfer phenomena and by scatter in the experimental data.

Zuber and Tribus[2,3] have found a more understandable situation in the transition-boiling case. Since the transition-boiling region joins with the nucleate-boiling region at the point of maximum heat flux, this approach is a practically useful one for attacking the problem of burnout. Because the situation analyzed is that in which the liquid does not touch the heat-transfer surface, effects of surface conditions do not enter.

The case that has been considered in detail is that of "pool boiling:" a situation in which the heat-transfer surface is a flat, horizontal surface with the water above it. Attention is focused on the situation in which a steam layer exists between the solid heat-transfer surface and the liquid. The boundary between the steam and liquid is unstable in a manner referred to as "Taylor instability."[4] Bellman and Pennington[5] have demonstrated that disturbances of such a surface can be stable or unstable, according to whether the wave length of the disturbance is shorter or longer than a critical value. The critical wave length is given by

$$\lambda_{01} = 2\pi \left[\frac{\sigma}{g(\rho_L - \rho_V)} \right]^{1/2}$$

where σ is the surface tension, ρ_L is the density of the liquid, ρ_V is the density of the vapor, and g is the acceleration of gravity. It has further been shown that the wave length for which the amplitude of the disturbance grows most rapidly is given by

$$\lambda_{02} = 2\pi \left[\frac{3\sigma}{g(\rho_L - \rho_V)} \right]^{1/2}$$

The instabilities of the surface take the form of spikes of the liquid which move downward toward the solid surface. It has been shown by Taylor[4] that the initial growth of the amplitude of such disturbances is exponential in time. It is postulated by Zuber and Tribus[2] that the spikes of liquid never actually touch the solid surface because they exist in the spheroidal state: the state in which the liquid is prevented from touching the hot surface because of the formation of a vapor layer next to the surface by evaporation from the liquid. The example of the spheroidal state, which is often quoted, is that of droplets of water on a hot plate; they will dance over the surface of the plate without actually touching it, while slowly evaporating. Most of the heat transfer associated with the transition-state boiling is assumed to occur in these close approaches of the liquid to the hot surface. The characteristics of the transition-boiling region to be expected from this picture

are summarized in the following quotation from reference 2:

> 1. As a consequence of Taylor instability a definate geometrical configuration can be expected in transition boiling. For a two-dimensional system this geometry is characterized by disturbances with wave length in the spectrum
>
> $$\lambda_{01} \leq \lambda_0 \leq \lambda_{02}$$
>
> 2. As a consequence of the exponential character of the instability it can be expected that the interface takes the form of spikes of heavy liquid moving downward and of rounded regions of lighter fluid moving upwards.
>
> 3. It can be expected that, because of the 'spheroidal state,' no solid-liquid contact exists in transition boiling. The change from transition to nucleate boiling corresponds to the collapse of the spheroidal state.
>
> 4. The temperatures of the solid surface corresponding to the termination of nucleate boiling and of transition boiling do not necessarily coincide. Therefore, at the peak flux nucleate boiling and transition boiling can simultaneously co-exist.
>
> 5. As a consequence of the definite geometrical configuration a release of bubbles from the interface at regular (spatial) intervals can be expected. For large evaporation rates the release of a bubble will appear as a 'burst.'
>
> 6. Because of the release of vapor and of Taylor instability the process exhibits, also, a periodicity in time. The phenomenon is hydrodynamically unstable but thermally stable.
>
> 7. Inasmuch as the factors which influence the geometry remain invariant, it can be expected that, in transition boiling, changes in heat transfer rates are associated with changes in frequencies only. The maximum and the minimum heat flux correspond, therefore, to the maximum and the minimum allowable frequencies of the system.

Although the pool-boiling case was the one considered by Zuber and Tribus,[2] the concepts are applicable to other heat-transfer conditions. Such applications are discussed in reference 2, and expressions are derived which permit the prediction of the maximum nucleate heat flux in pool boiling of saturated or subcooled liquids. It is shown that the empirical constants involved in certain previous empirical correlations of peak nucleate heat flux can be derived by the theory. It is also shown that the observations of Westwater and Santangelo[1] are quantitatively consistent with the theory with respect to such characteristics as frequency of bursts of bubble formation and diameter of bubbles produced. The analysis also leads to the conclusion that because of the statistical nature of the disturbances responsible for the heat transfer and the bandwidth of the unstable wave lengths which govern the process, an inherent uncertainty exists in determining the exact value of the heat flux at burnout. The width of this uncertainty range is estimated as approximately ±14 per cent. Consequently, the poor reproducibility of experimental data on burnout is to be expected.

Although the approach of Zuber and Tribus[2] has yet to be applied to a large number of practical cases, it nevertheless appears to represent a promising method of approaching the problem of transition boiling and burnout.

Transient Heat Transfer

Reference 6 is a mathematical study of the transient heat transfer in a generalized reactor-coolant channel, applicable to forced-convection cooling of heterogeneous reactor fuel elements. No reactivity interaction and no change of state of the coolant fluid are considered. The mathematical formulation and solutions should be useful for analysis of transient heat transfer in specific reactors. Impulsive, step, linear, and step-exponential variations of inlet temperature are considered, as well as uniform impulsive and step variations of heat generation.

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Stability of Boiling-water Reactors

Since boiling reactors were first proposed, it has been recognized that stability has been the major feasibility question and that its limitations might very well determine the maximum power density achievable in such reactors. The early BORAX reactors demonstrated the feasibility of the concept, but they were limited in power by the occurrence of unstable oscillations. These first reactors were designed to encounter the stability problem at low power densities in order to minimize the costs of the feasibility tests. The construction and operation of later boiling-water reactors—EBWR, BORAX-IV, and the Vallecitos Boiling Water Reactor (GE)—have demonstrated that the power limitations set by stability are considerably higher than those which applied for the early BORAX reactors. The excellent power performance of these reactors and the limitations on heat output from fuel elements which have resulted from the adoption of oxide fuel elements for water-cooled reactors suggest that in the future the power-density limitation in boiling reactors may be set by temperature-drop limitations in the fuel elements rather than by stability considerations. In such a case the maximum power density attainable in a boiling-water reactor will be little different from that attainable in a nonboiling-water reactor that is also fueled with oxide elements. There are, of course, other design limitations on the boiling reactor, notably the requirement of attaining the proper steam-void reactivity coefficient, which puts certain limitations on the fuel-to-moderator ratio that may be used. However, a comparison of current designs of boiling- and nonboiling-water-cooled reactors indicates that the fuel-to-moderator ratios used are not greatly different, and this restriction probably will not have a large effect on the attainable power density.

A number of papers have recently been published dealing with the stability of boiling-water reactors. The following discussion is based on those papers and on the earlier results of the BORAX experiments.

At the time the boiling-water reactor was first proposed, it was recognized that such reactors would probably have to operate with relatively large amounts of reactivity compensated by steam if economically attractive power densities were to be attained. Theoretical analyses of stability, based on the best models of steam formation and steam flow which could be constructed at that time, indicated there was little chance that the reactors would be unstable as long as no reactivity fluctuations occurred which approached the magnitude of the delayed neutron fraction β. In these early stability models the time lag in transferring heat from the fuel elements to the water and the time delay in removing steam from the reactor were taken into account. Hydrodynamic motions involving the inertia of the water in the reactor were not considered. It was quite evident that the power of the boiling reactor would be self-regulating, but the analyses did not and could not predict the effectiveness of power regulation because little was known about the transient production and flow of steam. In the language of the servo engineer, there was no way of knowing how noisy the regulating system would be.

From another point of view, the situation might be described as follows: It was expected that the boiling reactor would have to operate with a steam content that represented a relatively large amount of negative reactivity. If the steam content of the core could change by a large fraction, then evidently the reactor would be subjected to a large amount of excess reactivity which might be dangerous. There was no obvious mechanism to cause the sudden disappearance of steam from the core, but neither was there reason for assurance that the rates of steam production and flow out of the reactor would behave consistently. Observation of the boiling phenomenon would lead to the belief that it might be subject to rather large fluctuations. However, the calculations indicated that, if there were no large time lags in the formation of steam at the fuel-element

surface, the self-regulating property of the reactor would be sufficiently effective to maintain control in spite of reasonably large fractional fluctuations in steam content. This expectation and the analyses which indicated stability, provided reactivity fluctuations did not exceed the delayed neutron fraction, gave some measure of confidence that the boiling reactor would operate up to some void fraction which, in terms of reactivity, was of the order of the delayed neutron fraction. It seemed probable that the limit might well be a few times the delayed fraction.

When the BORAX-I experiments were begun, the first characteristic investigated was the reactor's ability to protect itself against sudden increases of reactivity. In a series of experiments, progressively greater amounts of excess reactivity were applied, up to a maximum of about 2.1 per cent, and the reactor demonstrated its ability to protect itself against the effects of the resulting power excursions. It was then quite evident that the reactor could be safely operated with any amount of reactivity up to 2.1 per cent compensated by steam voids. The reactor was then operated as a steadily boiling reactor at atmospheric pressure, with steam-void reactivities[1] up to about 2.5 per cent. The expected irregular fluctuations in reactor power associated with the boisterousness of the boiling phenomenon were observed. Their amplitudes were not large, and so far as could be ascertained, they never led to reactor instability. Instability was observed when the amount of reactivity held by steam voids was made sufficiently high, but it did not appear to be connected with the irregular power fluctuations. There was a range of power just below that at which instability occurred, in which regular oscillations, of definite period, were observed. When instabilities did occur, it was always through the progressive increase in amplitude of these oscillations over several cycles. Later BORAX experiments showed that the attainable power density from the reactor could be increased if the reactor were designed so that the steam-void coefficient of reactivity was smaller and that an increase in operating pressure improved stability markedly.

Aside from the simple demonstration of boiling operation, the early experiments demonstrated three characteristics which were of fundamental importance to the concept of the boiling reactor and which, in fact, constituted the major assurance that the boiling reactor had practical promise. These characteristics were

1. The time lag associated with the formation of steam at the fuel-plate surface was sufficiently short to be practically negligible in determining the characteristics of the steady-boiling reactor. This characteristic in effect guarantees that there can be no large-scale collapse of steam bubbles so long as the reactor continues to produce power.

2. The behavior of the reactor in both transient and steady-state tests was consistent and repeatable. This characteristic gave assurance that the gross behavior of the steam in the reactor core was not subject to fluctuations beyond the control of the reactor designer.

3. The random power fluctuations present during steady-state operation were not of serious magnitude. This characteristic demonstrated that the local fluctuations in steam density were not of sufficient magnitude, even in the rather small BORAX reactors, to constitute a serious barrier to the steady operation of the boiling reactor.

In combination these results indicated that the behavior of the reactor should be understandable in terms of the neutron kinetics of the chain reaction and the kinetics of the gross hydraulic and heat-transfer processes of the steam.

Although the BORAX reactors were not equipped to give information as to the cause of the oscillation which eventually led to instability, the behavior was suggestive of a hydraulic oscillation. At the same time it was obvious that energy was being fed into the oscillation through a coupling with the reactor power. If the steam had been formed by a steady source of heat, the same type of oscillation might have occurred, but its amplitude would have been limited. One means of approaching the characteristics of a constant heat source would be to use fuel elements of high heat capacity and low thermal conductivity, rather than the thin aluminum fuel elements used in the first three BORAX reactors. Such fuel elements would tend to smooth out the effects of fluctuations in reactor power and introduce a damping in the feedback from power fluctuation to steam production. They would at the same time, of course, reduce the capability of the reactor to protect itself against sudden reactivity increases. Aside from the question of stability, such fuel elements become almost a necessity when large power reactors are considered because the elements of low thermal time constant

can only be made from highly enriched uranium. The fuel elements for the EBWR, consisting of relatively massive plates of uranium metal clad with zirconium, have a time constant of approximately ½ sec as compared to about 1/10 sec for the early BORAX fuel elements.

In order to study the characteristics of fuel elements having long thermal time constants, with greater freedom from operating restrictions than is possible with EBWR, such elements were made for the BORAX installation and constitute the reactor known as BORAX-IV. These elements are composed of cylindrical slugs of thorium and uranium oxides in aluminum jackets, bonded to the jackets with lead. Although the oxide mixture has a much lower thermal conductivity than uranium metal, the thermal time constant of these elements is only roughly twice that of the EBWR elements because the diameter of the slugs is small. Both BORAX-IV and EBWR have exhibited much improved stability over the early BORAX reactors. In the case of BORAX-IV this must be attributed entirely to the different fuel element. The prompt negative temperature coefficient due to Doppler effect, as well as the longer thermal time constant, may have an effect. In EBWR, the higher operating pressure and possibly other factors are also involved. The Vallecitos reactor is reported to demonstrate excellent stability also. The improvement is primarily due to higher operating pressure, 1000 psi, since it uses fuel elements of short thermal time constant.

As a result of the experience which has been gained with boiling reactors, there is no longer a fear of some unknown mechanism which could cause a collapse of steam bubbles or a sudden unexpected transition from stable to unstable operation. The question of boiling-reactor stability is recognized as a complex one but one which is amenable to understanding and analysis. The recent papers which have been written on the subject do much to clarify the behavior which has been described above and to reduce it to a quantitative basis. The paper by Weil[2] is a summary of some of the recent work and will be discussed briefly here.

The first point made is that the relative importance of the various reactivity feedbacks varies so greatly between a reactor operating at atmospheric pressure, as the early BORAX reactors did, and a reactor operating at the elevated pressures characteristic of power reactors that they cause an effectively qualitative difference between the behavior of the two reactor types. It is further argued that instabilities observed at low pressure do not necessarily mean that reactors operating at high pressure would exhibit similar instabilities. Theoretically, the very great difference originates in the very much greater sensitivity of the reactivity of the low-pressure reactor to changes in pressure. Weil demonstrates that for a typical boiling reactor operating at atmospheric pressure, the pressure coefficient of reactivity (Δk) may be as high as 37 per cent per pound per square inch, whereas, for a similar reactor operating at 600 psi, it would be only 0.07 per cent per pound per square inch. In these calculations the slip of steam through water is neglected. Such a large pressure coefficient of reactivity in the low-pressure reactor would constitute a very strong coupling between the pressures developed by inertial and gravitational forces in any hydraulic oscillations which may occur and reactivity. The pressure coefficient of reactivity is a positive one and would therefore represent a method of feeding energy into the hydraulic oscillations. It is argued that the instabilities observed at atmospheric pressure are probably due to such hydraulic oscillations and that this type of oscillation, although overwhelmingly important in the low-pressure reactor, would be only one of several small feedback effects operating on the high-pressure reactor. Two possible types of hydraulic oscillations are recognized: one is simply an oscillation of the water surface such as might occur if a tub of water were rocked back and forth; the other is an oscillation in which the gravitational and inertial forces on the water act against the compressibility of the steam. The first type of oscillation is referred to as the U-tube type; the second is referred to as the bouncing-piston type.

A question that might be raised is the validity of neglecting the effect of the slip of steam through water. It is known that at atmospheric pressure the slip velocity is quite large. A surprising result of the early BORAX experiments was the relatively high value of power obtained at atmospheric pressure for a given fraction of steam void in the reactor. Later laboratory experiments showed that this characteristic was due to a very high slip velocity between the steam and the water. As pressure is increased, the slip velocity decreases, and

at pressures in the 600-psi range the relation between void content and power output approaches something like that which would be calculated on the assumption of no slip. The result is that the pressure coefficient of reactor power is very much lower than would be calculated on the assumption of no slip. The measurements on power output were steady-state measurements and do not necessarily apply to the transient case which is involved in power oscillations. Nevertheless, they do indicate that slip effects can be large. The question is not, however, one of extreme importance since the effect of slip is to improve the situation in all cases. Thus it is not expected that the high-pressure cases will be worse than the theory indicates but rather that both the high- and low-pressure cases may be better and that the low-pressure cases may be considerably better. The more important fact is that a very marked increase in stability was experimentally observed as operating pressure was raised. The theory gives at least a qualitative explanation for the observation.

More recent experimental information on the effect of pressure was obtained in connection with the program to increase the power of EBWR.[3] In this program the transfer function of the reactor[4,5] (i.e., the frequency response of the reactor power to sinusoidal variations in reactivity) was measured under various operating conditions. The frequency response at one-fourth power and operating pressure was shown to be very nearly the same as that at full power and full operating pressure. It was found possible to predict the behavior at higher powers and full pressure by observing the behavior at rated power but at reduced pressure. These effects appear to be caused primarily by the variation of the ratio of steam-void content to power output with pressure and are in fact an indication that the slip velocity of steam through water does not change markedly over the pressure ranges investigated. Such variations in pressure by factors of 2 to 4 are of course much less drastic than those represented by a change from atmospheric pressure to, e.g., 600 psi.

Having made the point that the high-pressure power reactor is in effect qualitatively different in its behavior from the atmospheric pressure reactor, Weil[2] considers the various factors which enter into the reactivity feedback loop in the high-pressure reactor. These effects can best be recognized by assuming that the reactivity of the reactor is suddenly increased by some outside means. Such a reactivity increase will tend to increase the reactor power in a manner determined by the neutron kinetics of the chain reaction. The increased power will be transferred from the fuel elements to the water but with a time lag which is characteristic of the thermal time constant of the fuel elements. The increased rate of steam generation will require that additional water be displaced from the reactor core. Involved in this transient displacement will be the inertia of the water, the elastic constants of the steam, the gravitational forces on the water, and the damping forces due to the flow resistances in the circuit. Evidently the resulting motion of the water might involve one or more characteristic modes of oscillation, each of which would produce an effect on steam-void content of the core, with a characteristic time constant. The quantitative relations of these various responses can be studied by means of the frequency response methods that are normally employed in servo engineering. The usual stability criteria can be employed to specify the conditions under which the system will be stable or unstable.

The general conclusions which may be drawn from the discussion of Weil[2] are as follows:

1. The low-pressure coefficient of reactivity of a boiling reactor operating at high pressure reduces the positive feedback effects which may be connected with hydraulic oscillations of the water to a negligible magnitude. Consequently, reactors at high pressure are not to be expected to become unstable from hydraulic oscillations.

2. The possibility of hydraulic oscillations can be further reduced by suitable damping in the recirculation system.

3. The gain of the feedback system can be reduced by the use of fuel elements with long thermal time constants. If the time constant of the fuel element is well separated from the time constant inherent in the system hydraulics, the system may be stabilized to a large degree. Such long time constants tend to result from the use of oxide fuel elements. It is, of course, true that the time constants will become shorter if the fuel elements are made smaller in order to allow higher power densities without exceeding the temperature limits of the oxide. Nevertheless the excellent behavior of BORAX-IV, using small diameter oxide elements, is very encouraging.

4. The dynamic characteristics of the boiling reactor are understood, at least in principle.

The design principles that promote stability are recognized. Although it is not yet possible to predict the stability limits of the boiling reactor accurately from first principles because of the complexity of the physical situation, it is possible with the aid of oscillator techniques to map out completely safe operating ranges for the reactor after it has been built. The difficulties in predicting exact limits of stability during design of the reactor may lead to overconservatism in the design, but the experimental indications are that factors other than stability are likely to be limiting on power density in reactors fueled with uranium oxide.

Transfer-function Calculations

The use of the concept of the reactor transfer function (see pp. 10-14) for both the theoretical and the experimental analyses of reactor dynamics has become rather general. Recent reports that have application in this field are references 6, 7, and 8. References 2, 3, and 5 also employ this approach, and a good discussion of the application of the concept is given in reference 9.

Reference 6 is a derivation of the transfer function for a water-boiler (i.e., low-power aqueous homogeneous) reactor, which includes the effects of delayed neutrons, temperature, and radiolytic gas production. Reference 7 is a derivation of the transfer function for a reactor that has a constant average excess reactivity and whose average power is therefore rising on an exponential period of constant doubling time. As would be expected, the transfer function deviates from that of the critical reactor at low frequencies, reflecting the effectively lower delayed neutron fraction. Reference 8 describes a machine code, JUGGLER, for the IBM 704, whose function is to analyze the over-all power-coefficient data obtained by pile oscillator measurements to produce best values for the individual zero frequency power coefficients and their respective time constants.

Analyses of SPERT Tests

The SPERT program is a continuation of the program[10,11] for experimental investigation of the safety of water-cooled water-moderated reactors, which began with the BORAX experiments.[12,13] It is expensive and time-consuming to extend tests of this type to all conceivable cases which would be of interest for various reactor types. The tests that have been made to date apply directly to reactors using fuel plates of low heat capacity and high thermal conductivity. These are primarily research reactors and certain small power reactors. In order to make the results useful for more general cases, it is highly desirable that a quantitative theoretical understanding of the test results be obtained if possible. Needless to say the situation to be analyzed is a complex one, and it should not be surprising that a complete theory has not been developed. References 14, 15, and 16 summarize recent work which has been done on the theory of the behavior of the SPERT reactor. A number of useful relations are derived for specific aspects of the reactor behavior. These include the kinetics of the neutron chain reaction, theoretical studies of bubble formation, hydrodynamic and thermodynamic considerations, and oscillations of reactor power.

Fission-product Release from Molten Reactor Fuels

In solid fuel reactors the fuel element constitutes the first line of defense against the fission-product release. All serious reactor accidents postulated in such reactors involve first the melting of the fuel elements. A question of serious importance to the study of reactor safety is the question of which fission products, and in what percentages, are released from the molten elements. Evidently the answer to this question will depend upon the materials and design of the fuel element itself, the atmosphere in which the melting takes place, the maximum temperature reached, and other factors. These factors have been considered by Parker and Creek,[17] who have reported experimental measurements covering some of the conditions of interest. They point out that the following processes favor the release of a given fission product to the surrounding atmosphere.

1. Vaporization of the elemental form of a volatile constituent. This is particularly important for elements such as the rare gases, xenon and krypton, and for I_2 and Br_2, and cesium and rubidium.

2. Diffusion into an oxide, slag, or skin followed by dispersion or vaporization of the oxide at high temperature. This may occur with molybdenum.

3. High-temperature vaporization of the element in the absence of air, e.g., strontium, barium, selenium, and tellurium.

4. Formation of volatile chemical compounds between two fission products, particularly with halides.

The following processes are mentioned as favoring retention of the fission product in the melt:

1. Occlusion of elemental or combined forms of volatile constituents. The rare gases xenon and krypton are held by molten aluminum unless purged.

2. Conversion to nonvolatile carbides, e.g., those of zirconium and niobium.

3. Occlusion in crystalline fuel, e.g., UO_2 in a molten stainless-steel matrix.

4. Formation of stable alloys with the uranium fuel, e.g., tellurium and ruthenium.

5. Formation of nonvolatile oxides, e.g., BaO and SrO.

The following generalizations were made from the experimental melting of irradiated fuel-element specimens:

1. Prolonged heating of specimens in air above their melting points released a large portion (up to 50 per cent) of the radioactivity.

2. A moderate amount of heating in air or steam, sufficient only to cause melting, results mainly in the partial volatilization of iodine, bromine, cesium, rubidium, and the rare gases; in the presence of air or water vapor, strontium and other fission products were not released in any significant quantity.

Specific results for specimens of the APPR, MTR, and STR types are summarized below.

1. *Melting of APPR Type Elements.* Typical test results with slightly irradiated elements melted slowly at 1525°C in air showed a gross gamma-activity release of 8 per cent maximum. This was composed mainly of ~50 per cent rare gases, ~33 per cent iodine, ~9 per cent cesium, and a maximum of 0.2 per cent strontium. Results of similar tests at 1600°C on APPR specimens that had been irradiated to a burn-up of 25 per cent showed a maximum gross gamma-activity release of ~18 per cent, with up to 70 per cent of the cesium being released and no strontium. When specimens were melted in a helium atmosphere, the gross-activity release was similar to the above case, but the amount of released strontium increased to ~6.8 per cent, and virtually all the cesium was released. This difference in behavior is attributed to the fact that in the presence of air or water vapor the alkaline-earth metals, barium and strontium, form oxides which do not dissociate readily. When heated rapidly to above 1720°C, the APPR specimens showed gross gamma-activity releases of up to 52 per cent.

2. *Melting of MTR Type Elements.* Slow melting of MTR elements at 700 and 800°C indicates a release of up to ~1 and 2.5 per cent, respectively, of the gross gamma activity, with a maximum release of up to 2 per cent iodine, 10 per cent rare gases, and negligible amounts of the other fission products. The behavior of this aluminum alloy is of interest since many of the research reactors located in populated areas and on college grounds use aluminum alloy type elements. The low amounts of activity and fission products released on simple meltdown of the aluminum alloy fuel are reassuring. The data, of course, can only be applied to the case of simple meltdown (i.e., the case in which the aluminum melts without reaching temperatures greater than 800°C and the melt remains undisturbed during cooling).

3. *Melting of STR Type Elements.* The STR zirconium alloy (melting point, ~1850°C) specimens having a burn-up of 15 per cent were melted in air, steam, and helium at approximately 1900°C. As a result of the high melting temperature, these specimens released 95 to 99 per cent rare gases, 40 to 80 per cent iodine, up to 66 per cent cesium in helium, up to 5 per cent cesium in air, and a trace (<½ per cent) strontium.

The experiments reported here evidently represent only a narrow range of cases which may be of interest. An extension of the results to a wider range of conditions would be of great interest to the reactor designer.

Containment

Three of the papers presented at the recent Reactor Safety Conference in New York City, Oct. 31, 1957, treated subjects related to the containment of fission products after an assumed reactor accident involving a substantial energy

release. Reference 18 considered the containment problem in general, reference 19 discussed the effects of pressure waves arising in a reactor explosion, and reference 20 dealt with the cost of containment and other safety provisions.

Brittan[18] pointed out the requirements for the structure which is usually referred to as the secondary containment vessel, or the containment envelope, and pointed out areas in which present knowledge is deficient. The containment envelope is usually a spherical or cylindrical steel tank that surrounds all the reactor parts that are radioactive or could become radioactive in an accident. Its function is to prevent release of fission products to the atmosphere in the event they are released from the normal boundaries of the reactor by an accident. To perform its function, the envelope must be sufficiently strong to withstand any pressures produced by the accident and it must be protected from any missiles originating in the accident which could puncture it. The total energy theoretically available in a hypothetical accident — the nuclear energy plus any energy released by chemical reactions such as zirconium-water reactions or sodium-air reactions, plus energy stored in the form of pressurized coolant — can be quite large. If it is all added together and assumed equal in effect to an equal energy release from TNT, the resulting pressure on the walls of the container may be so high that the requirements on the containment vessel become economically impractical. Brittan[18] points out that such a procedure for estimating the destructive effect of the accident may be a gross overestimate; a more realistic estimate must take into account practical limitations on the magnitude of the energy release; the time rate of energy release, which, in general, will be much slower than that from a chemical explosive such as TNT; and dissipation of the energy in the materials immediately surrounding the reactor.

Since the moderator, coolant, reactor vessel structure, and particularly the shielding usually represent rather large masses of material which are present around the source of energy, the dissipation of energy in them must be considered, and the possibility of designing the reactor and its surroundings to make maximum use of the material in this way should be considered.

After the pressure wave from the hypothetical energy release has been attenuated sufficiently to allow a reasonable design for the containment envelope, it remains to ensure that no missiles originating in the explosion can reach and puncture the envelope. All missiles except very large ones may be stopped by suitable barriers such as woven steel mats. Potential missiles of very large size, such as large shield plugs, must be anchored so that they cannot acquire appreciable velocities and that large fragments of them cannot be broken off and accelerated as missiles. An example of such anchoring is the top plug of the EBWR reactor.[21]

The following deficiencies were mentioned as handicaps to progress in developing effective and economical containment designs.

1. Realistic criteria for permissible leak rates from containment structures have not been established.
2. Good methods of proving the adequacy of containment, particularly on a continuing basis after reactor start-up, have not been developed.
3. A separate code for the design of unfired vessels having a once-in-a-lifetime loading is needed.

Porzel,[19] in his discussion of pressure waves resulting from hypothetical accidents, concluded that the most effective means of containing the energy of a reactor explosion appears to be by blast shields which absorb energy by crushing the material of the shields. It was pointed out that a good blast shield may consist of several regions of different crushing strength, the regions of lower strength being used farther out to provide good energy absorption as the pressure wave is attenuated. Channeling of the pressure wave can be used to advantage in certain cases, particularly to protect portions of the structure which must stay intact, such as the structure provided to hold down large potential missiles.

Second in importance only to the question of how to provide adequate containment is the question of its cost. Because containment criteria have varied markedly even over the relatively short history of power reactors, a cost analysis based on experience shows wide variations — from 3.5 to 21 per cent of the total plant costs. Size of the installation also has a large effect, and the high fractional costs for containment apply to small plants. It is reported that a study was made for ANL, by the engineering firm of Sargent & Lundy, of two boiling reactor plants, each with and without containment, for several containment arrangements. One of the reactors was small, having a heat output of 20 Mw,

whereas the other was large, having a heat output of 600 Mw. The containment included a hold-down structure, blast shield, and missile protection as well as the containment envelope. The cost for the small plant was 17.5 per cent of the total plant cost, or $40 per kilowatt of heat; for the large plant the containment represented 7 per cent of the total plant cost, or $6.50 per kilowatt of heat.

In a separate paper Kallman and Hanson[20] discussed the cost of containment and other special safety equipment to prevent large-scale accidents or to reduce their effects. Their discussion is centered around the Indian Point plant which is under construction for the Consolidated Edison Company of New York. This plant will contain a pressurized-water reactor having a heat capability of 585 Mw. The total electrical capability is to be 275 Mw, of which 163 Mw will come from the reactor and 112 Mw, from two oil-fired superheaters. The cost estimates presented were developed primarily in studies connected with this reactor. They are regarded by Kallman and Hanson as reasonably representative of those for large pressurized-water reactors in the United States. The special safety equipment is summarized in Table III-1, along with rough cost estimates that are based on a reactor of thermal output in the 500-Mw range. The items include the secondary containment sphere and items associated with it; shielding, which would be necessary only in the event of a release of fission products inside the sphere; extra transmission line, which might be necessary because of a semiremote location of the reactor (assumed to be about 30 miles from the optimum economic location for a power plant); and extra auxiliary equipment connected with shutdown cooling, emergency reactivity shutdown mechanisms, extra instrumentation for such shutdown devices, and other special items.

Table III-1 SPECIAL SAFETY EQUIPMENT AND COST[20]

Containment sphere	$1,500,000
Items associated with sphere	2,300,000
Excavation	
Foundation	
Structural steel	
Containment cooling system	
Ventilation system	
Spray system	
Pressure regulating system	
Extra shielding around sphere	1,200,000
Extra transmission line*	2,000,000
Extra auxiliary equipment	1,400,000
Auxiliary power supplies	
Shutdown cooling	
Emergency poison injection system	
Extra instrumentation channels	
Other special items	
Total	$8,400,000

*When plant location is more remote than that of a conventional plant.

It is estimated[20] that the cost of a conventional building to house the reactor, with no special provision for fission-product containment, would be about three million dollars. Consequently, the cost in Table III-1 attributable directly to safety should be reduced by three million dollars. When this is done, the net contribution of the special safety equipment items to the cost of the plant is five million, four hundred thousand dollars. For a plant having an electrical output equal to that of the Indian Point plant (163 Mw), the contribution of this expense to the cost of the plant would be $33 per electrical kilowatt of capacity.

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Isotope Build-up in Recycled Fuel

When the recycling of fissionable isotopes through nuclear reactors is considered, the build-up of the higher isotopes must be taken into account. The recycling of plutonium is necessary if an appreciable fraction of the energy potentially available in U^{238} is to be used. If the energy potentials of thorium are to be tapped, U^{233} must be recycled. The recycle of U^{235} to some degree is necessary for economic operation on the partially enriched U^{235}-U^{238} cycle, although the build-up of higher isotopes can be greatly reduced if the uranium is not recycled through the diffusion plant but brought back to a usable enrichment by the addition of highly enriched U^{235} separated from virgin natural uranium.

Two effects of the higher isotopes are of importance: (1) the effect on neutron economy and (2) the effect of their radioactivity on the handling difficulties which arise in reprocessing and refabricating the fuel. Both of these considerations will vary importantly with the details of the irradiation and recycling processes. The quantities of the isotopes and their radiation levels can, however, be computed for various assumptions as to the characteristics of the processes. This has been done in references 1 and 2. Reference 1 treats the build-up of the isotopes originating in the irradiation of U^{235}. The resulting isotopes of importance are U^{236}, U^{237}, Np^{237}, and Pu^{238}. As an example, the paper lists the effects that would occur in reactor fuel containing 2 per cent U^{235} which has been irradiated to 4000 Mwd/ton in each pass through the reactor and has been recycled until steady-state conditions are reached. If U^{236} is not removed by the diffusion plant, the following effects occur:

1. The neutron poison due to U^{236} could amount to as much as 100 to 150 barns per U^{235} atom.

2. The build-up of U^{237} could increase the required uranium product decay time from the approximately 100 days necessary for present-day fuels to approximately 160 days for an average power reactor fuel.

3. The chemical separation plant may have to separate Np^{237} as well as the fission products. The Np^{237} may be present to the extent of 1 kg/ton.

4. The build-up of Pu^{238} will contaminate the plutonium product and increase the difficulty of handling it. High Pu^{238} concentrations will increase the biological hazard associated with plutonium and may also increase the neutron background by the α,n reaction unless a high degree of decontamination from light elements is obtained.

It is stated that the removal of approximately 25 per cent of the contained U^{236} per pass through the diffusion plant will reduce all these effects by at least an order of magnitude.

Reference 2 treats the build-up of heavy isotopes from the irradiation of uranium, plutonium, and thorium. A large number of useful results are given in the form of graphs. A rough idea of the effects of higher isotopes on the radioactive levels of irradiated fuels, decontaminated from fission products, can be obtained from Table IV-1, which is taken from reference 2. The table gives the approximate activity level at 1 ft from a 1-kg sphere of the fuel in question.

Table IV-1 APPROXIMATE ACTIVITY LEVEL OF A 1-KG SPHERE IN AIR[2]

Material	Activity mr/hr at 1 ft
Natural U in secular equilibrium with daughters	0.2
Natural Th in secular equilibrium with daughters	1.1
U^{235} in secular equilibrium with daughters	3.9
Pu with 9% Pu^{240} from single 100-day U irradiation at 5×10^{13} n/(cm^2)(sec)	108
U^{233} from 3000 g/ton Th after 2-year storage (2–13 mb Th n,2n cross section)	750–4850
Recovered Th from a single irradiation of 3000 g/ton processed 60 days after irradiation*	1.6

*1.68-kg slug, $\phi\sigma_{02}(n,2n) = 10^{-13}$.

Re-enrichment by Blending

In connection with the question of recycling enriched uranium fuel through the diffusion plant, a recent study[3] of the relative economics of re-enrichment by this method and by the addition of highly enriched U^{235} is of interest.

The study is based on the assumption of partially enriched UO_2 as the reactor fuel. The residue from the chemical processing of a spent reactor loading consists of uranyl nitrate of some enrichment lower than that required for the reactor. UO_2 of the required enrichment may be obtained via two alternative routes: (1) the uranyl nitrate may be converted into UF_6 and returned to the AEC diffusion plant for re-enrichment, or (2) sufficient fully enriched uranium (~95 per cent U^{235}) may be purchased from the diffusion plant, converted to UO_2, and blended with the UO_2 obtained directly from the nitrate to produce fuel of the necessary total enrichment. The relative economics of the two methods has been studied by Kallman and Brennan[3] using an assumed cost of $10 per kilogram of uranium for all conversion costs, and 1 per cent chemical loss in conversion processes. (Conversion of fully enriched UF_6 to UO_2 was assumed to have a unit cost proportional to the enrichment, or about $330 per kilogram of uranium.) The results of their comparison are shown in Table IV-2. Their conclusions were: that blending with highly enriched uranium may permanently be practiced in place of re-enrichment at the diffusion plant because the difference in cost may be small and the flexibility greater. This conclusion is based upon the assumed costs which, for illustrative purposes, were taken between the current commercial costs, and the cost levels expected in the future.

Since publication of reference 3, some cost figures for conversions have been set by the U. S. Atomic Energy Commission, particularly that for the conversion of uranyl nitrate from the chemical processing plant to the UF_6 form.[4] This cost has been set at $5.60 per kilogram of contained uranium for enrichments of 5 per cent or less and $32 per kilogram of contained uranium for enrichments greater than 5 per cent. On the basis of an available commercial price for conversion of UF_6 to UO_2 of $7.72 per kilogram per per cent enrichment (1.5 per cent greater), the comparison at present-day costs for the same enrichments would be as given in Table IV-3.

Table IV-2 COST OF 1 KG OF 3 PER CENT ENRICHED URANIUM IN OXIDE FORM[3]

Re-enrichment*	
3% enriched U	$375.50
2% enriched U	220.00
Difference	155.50
Conversion, nitrate to UF_6	10.00
Loss during conversion	2.20
Conversion, UF_6 to UO_2	10.00
Loss during conversion	3.76
Total	$181.46
Blending†	
10 g fully enriched U	$171.00
Conversion, nitrate to oxide	10.00
Loss during conversion	2.20
Conversion, 10 g UF_6 to UO_2	3.30
Loss during conversion	1.71
Total	$188.21

*From 2 to 3 per cent U^{235}.

†Blending fully enriched U^{235} with 2 per cent to obtain 3 per cent.

Table IV-3 PRESENT DAY COST OF 1 KG OF 3 PER CENT ENRICHED URANIUM IN OXIDE FORM

Re-enrichment	
3% U^{235}	$375.50
2% U^{235}	220.00
Difference	155.50
Conversion, nitrate to UF_6	5.60
Loss during conversion	2.20
Conversion, UF_6 to UO_2	23.16
Loss during conversion	3.76
Total	$190.22
Blending	
10 g fully enriched U	$171.00
Conversion, nitrate to oxide*	5.60
Loss during conversion	2.20
Conversion, 10 g UF_6 to UO_2	7.72
Loss during conversion	1.71
Total	$188.23

*Assuming conversion of uranyl nitrate to oxide at the same unit cost as conversion of uranyl nitrate to UF_6.

In the practical situation the fully enriched UF_6 would be converted to nitrate and blended at that point of the process (probably at a somewhat lower cost) rather than being converted separately to the oxide.

From cost evaluations presented in Tables IV-2 and IV-3, it would appear that under present-day costs it would be less expensive to blend fully enriched U^{235} with residual fuel recovered

from the separation plant and that the advantage would decrease or shift to re-enrichment as conversion costs are decreased. However, a more detailed study would include blending and analyzing costs, and these might swing even the present day advantages to re-enriching. Also, it should be pointed out that this estimate (1) does not consider the effect of use charges and times required for the conversion steps, (2) is computed for a single enrichment, (3) assumes a cost of conversion of uranyl nitrate to UO_2, and (4) does not consider the possibility of a reduction in chemical reprocessing charges.

If economic methods of refabricating radioactive fuels are developed, the blending scheme may still have advantages; the blending of fully enriched uranium with contaminated uranium could then be considered. For return of uranium to the diffusion plant, the spent fuel must be decontaminated of fission products and converted to the UF_6 form. However, if contaminated UO_2 could be pelletized and fabricated into fuel elements, a considerable saving might be possible by using a simple chemical reprocessing of spent fuel elements (such as uranyl peroxide precipitation followed by hydrogen reduction) to recover UO_2 in a partially contaminated form. This material could be blended with fresh fully enriched uranium from the diffusion plant to obtain reusable fuel of the desired enrichment. Hence the blending method potentially might offer a larger reduction in fuel costs than the re-enrichment method.

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In previous issues of this Review, many of the current results of fuel and fuel-element developments and tests have been summarized.[1–3] The objective has been to make the summaries from the reactor designer's point of view. The reader is referred also to *Reactor Core Materials*, a companion publication in the Technical Progress Reviews, which is prepared from the point of view of the materials developer. It has seemed useful to attempt to consolidate the portion of the results which have bearing on those fuel elements which are actually in use or which can be considered to be reasonably well proved with respect to feasibility.

Elements are in use in power reactors employing three different coolants: H_2O, CO_2, and sodium. In the water-cooled reactors there are in operation two types of partially enriched elements (zirconium-jacketed UO_2 pellets, Shippingport PWR, and zircaloy-clad uranium-metal plates, EBWR) and two types of highly enriched elements (Zircaloy-clad zirconium-uranium alloy, Shippingport PWR, and stainless-steel–clad steel-UO_2 matrix, APPR). In addition to these it may be presumed that the element composed of a mixture of ThO_2 and highly enriched UO_2 is feasible. An element of this type has been used experimentally in the BORAX reactor and is contemplated for the Indian Point reactor.

In the gas-cooled reactor the only element that has seen service is the Calder Hall type, a uranium-metal slug in a finned magnesium alloy jacket. On the basis of the extensive UO_2 development that has been done for the water reactor and the well-established properties of stainless steel, it would seem reasonable to presume that the steel-jacketed UO_2 element that has been proposed in recent studies of gas-cooled reactors (pp. 53-58) would be also a feasible type.

In sodium-cooled reactors the element type in use is that consisting of uranium-metal slugs or rods jacketed with stainless steel, with a thermal bond of sodium between the uranium and the jacket. Such elements have been in use for a

(Text continues on page 27)

Table V-1 SOME PROPERTIES OF UNALLOYED URANIUM OXIDE[1–5]

Crystal structure, face centered cubic (CaF_2)
Theoretical density, 10.97 g/cm^3
Density (available commercially), 10.4 g/cm^3
Theoretical density (fissionable and fertile), 9.7 g/cm^3
$\frac{\text{Metal atom/cm}^3 \text{ in oxide}}{\text{Metal atom/cm}^3 \text{ in metal}}$, 51.5%
Melting temperature (mean), 2760°C; 5000°F
Linear coefficient of thermal expansion, 11.2×10^{-6} per °C
Specific heat, 0.056 cal/(g)(°C), (0–200°C)
Thermal-conductivity coefficient:
100% of theoretical density,[7] 0.008 cal/(cm)(sec)(°C), (1000°C); 1.93 Btu/(ft)(sec)(°F), (1000°C)
When thermally cracked (95% of theoretical density),[9] 0.00415 cal/(cm)(sec)(°F) (>1500°C); 1.0 Btu/(ft)(sec)(°F) (>1500°C)
Tensile strength, 5000 psi
Modulus of elasticity, 25×10^6 psi
Index of refraction, 2.35

Table V-2 SOME PHYSICAL AND CHEMICAL PROPERTIES OF URANIUM[62]

Phase	Density,[63–65] g/cm^3	Temp., °C	Crystal lattice type[63,65]
Alpha	18.7 – 19.0*		Orthorhombic
	19.04†	25	
	19.05 ± 0.02‡	25	
Beta	18.11	720	Tetragonal
Gamma	18.06	805	Body centered cubic

Transformation point	Transformation temp., °C: Heating	Cooling
Alpha-beta	665.6	656.7
	671.9§	662.6§
Beta-gamma	771.1	766.5
Melting point	1129.8	1129.6
	1131.80 ± 0.4§	1133.0 ± 0.4§
Boiling point	3820	

Mean thermal expansion coefficient for as-rolled alpha uranium,[67] 11.4×10^{-6}/°C
Thermal conductivity,[68] 0.08 cal/(cm)(sec)(°C); (400°C)
Tensile strength,[45] 90,000 psi (20°C)
Specific heat,[62] 6.649 cal/(mole)(°C) at 27°C; 8.952 cal/(mole)(°C) at 427°C; 11.107 cal/(mole)(°C) at 627°C
Heat of fusion,[69] 4.7 kcal/mole
Heat of vaporization,[69] 106.7 ± 0.1 kcal/mole

*Wrought material.
†Calculated from lattice parameters.
‡High-purity uranium, directionally solidified.
§See reference 66.

Table V-3 SOME PROPERTIES OF CANNING OR CLAD MATERIALS*

Density, g/cm³	Melting point, °C	Thermal conductivity: k, cal/(cm)(°C)(sec)	Thermal conductivity: t, °C	Thermal-neutron cross sections (2200 meters/sec): σ_a, barns	Thermal-neutron cross sections (2200 meters/sec): Σ_a, $cm^{-1} \times 10^3$	Linear thermal expansion: α, in./(in.)/(°C) × 10^6	Linear thermal expansion: t, °C	Yield strength (0.2% offset), psi × 10^{-3}	Tensile properties†: Tensile strength, psi × 10^{-3}	Tensile properties†: Elongation, %	Tensile properties†: Test temp., °C	Tensile properties†: Modulus of elasticity, psi × 10^{-6}
Aluminum – At. No. 13, At. Wt. 29.98												
2.7	660	0.503	100	0.230	13.9	28.7	0–600					
		0.530	200									
		0.546	400									
Aluminum – 1100-0 (annealed)												
2.72	643–657	0.53	25	0.221	13.3	25.6	0–300	5.0	13	45	24	10
								3.5	6	65	204	9
								1.0	2	85	371	4
Aluminum – 6061-T6 (hardened)												
2.70	582–648	0.37	25	0.233	13.9	25.4	0–300	40	45	17	24	10
								15	19	28	204	9
								2	3	95	371	4
Magnesium – At. No. 12, At. Wt. 24.32												
1.74	650	0.376	20	0.063	2.72	29.9	0–500	14	27	16	20	6.5
Magnesium – Magnox A-12												
1.74	650	0.276	0	0.064	2.5	27.9	20–400	16	20	8	20	6.5
		0.299	200					9.5–14	16	20	100	
		0.323	400					~0.7	~1.5	~80	400	
Zirconium – At. No. 40												
6.44	1845	0.054	100	0.180	7.65	5.69–6.15	0–600	32	54.5	40	27	13.7
		0.045	300					17	36	50	150	
		0.03	500					7.8	18.8	55	345	10.7
								5.2	11.2	88	500	9.2
Zirconium – Zircaloy-2												
6.45	1845	0.04	100	0.194	8.2	6.5	0–700	42	54.7	30	27	13.7
								17.5	23.5	40.4	345	10.7
		0.03	500					13.8	19.5	42.1	500	9.2
Stainless Steel Type 304												
7.92	1425–1510	0.035	100	2.99	219	18.7	0–650	30	80	50	25	28
		0.050	500					19	56	45	500	
Niobium – At. No. 41, At. Wt. 92.91												
8.4	2415	0.125	20	1.1	59.9	7.1	20	31.5	38	16	20	12.4
		0.140	300			7.3	300	25.5	36	~22	300	8
		0.156	600			7.55	600	20.5	28	40	600	4.7
Vanadium – At. No. 23, At. Wt. 50.96												
6.1	1900	0.074	100	4.98	306	10.4	23–900	64	72	28	24	20
		0.088	500									

*Data obtained from references 53, 69, and 73 to 75.

†Values are for fully annealed specimens unless indicated.

Table V-4 FUEL ELEMENTS USED IN NUCLEAR ELECTRIC POWER PLANTS COOLED BY WATER

Shippingport Pressurized Water Reactor (PWR)

Power: thermal, 225 Mw; electrical, 67 Mw (gross)
Coolant temp.: inlet, 509°F; outlet, 537°F
Fuel element (blanket): type, cylindrical; fuel, UO_2 pellets (0.3494 in. O.D.); enrichment, natural; clad, Zircaloy-2 (0.4130 in. O.D.); bond, helium
Fuel element (seed): type, plates; fuel, U-Zr alloy (2.05 × 0.039 in.); enrichment, 93%; clad, Zircaloy-2 (0.020 in. thick); bond, metallurgical

Yankee Pressurized Water Power Plant (PWR type)

Power: thermal, 480 Mw; electrical, 150 Mw (gross)
Coolant temp.: inlet, 487°F; outlet, 535°F
Fuel element: type, cylindrical; fuel, UO_2 pellets (0.30 in. O.D.); enrichment, 2.6%; clad, stainless steel (0.336 in. O.D.); bond, helium

Indian Point Pressurized Water Power Plant

Power: thermal, 500 Mw; electrical, 163 Mw (gross from reactor); 112 Mw (gross from superheater)
Coolant temp.: inlet, 480°F; outlet, 510°F
Fuel element: type, cylindrical; fuel, ThO_2–3 wt.% UO_2 pellets; clad, stainless steel; bond, helium

Army Package Power Reactor (APPR-1)

Power: thermal, 10 Mw; electrical, 2 Mw
Coolant temp.: inlet, 431°F; outlet, 450°F
Fuel element: type, plates (2.76 × 0.03 in.); fuel, UO_2 stainless steel dispersion (2.50 × 0.02 in.); enrichment, 93.2%; clad, stainless steel 304; bond, metallurgical

Dresden Nuclear Power Station (BWR type)

Power: thermal, 630 Mw; electrical, 200 Mw (gross)
Coolant temp.: outlet, 546°F
Fuel element: type, cylindrical; fuel, UO_2 pellets (0.500 in. O.D.); enrichment, 1.5%; clad, Zircaloy-2 (0.562 in. O.D.); bond, helium

Experimental Boiling Water Reactor (EBWR)

Power: thermal, 20–60 Mw; electrical, 5–15 Mw
Coolant temp.: inlet, 483°F; outlet, 488°F
Fuel element: type, plates (two sizes); fuel, U–5 wt.%–Zr–1.5 wt.% Nb (0.212 × 3.625 in.); (0.280 × 3.625 in.); enrichment, (1.44%) (natural); clad, Zircaloy-2 (0.020 in. thick); bond, metallurgical

BORAX-IV (experimental reactor)

Power: thermal, ~20 Mw; electrical, 2–4 Mw
Coolant temp.: outlet, 400°F
Fuel element: type, extruded tube sheet (tube, 0.225 in. O.D., sheet, 0.0875 in. thick); fuel, ThO_2–6.35 wt.% UO_2 pellets (0.225 in. O.D.); enrichment, 90%; clad, aluminum, M-388; bond, lead

Table V-5 FUEL ELEMENTS USED IN GRAPHITE-MODERATED ELECTRIC POWER REACTORS COOLED BY CO_2

Hinkley Point Plant (Calder Hall type)

Power: thermal, 980 Mw; electrical, 250 Mw (net)
Coolant temp.: outlet, 707°F
Fuel element: type, finned cylinders; fuel, uranium metal slugs (1.125 in. O.D.); enrichment, natural; clad, finned magnesium alloy; bond, mechanical, by reactor pressure

Kaiser-ACF (design only)

Power: thermal, 600 Mw; electrical, 215 Mw (net)
Coolant temp.: inlet, 473°F; outlet, 1000°F
Fuel element: type, tubular; fuel, UO_2 tubes, (0.75 in. O.D.), (0.32 in. I.D.); enrichment, 3%; clad, stainless steel 304 (0.79 in. O.D.); bond, helium

Table V-6 FUEL ELEMENTS USED IN NUCLEAR ELECTRIC POWER PLANTS COOLED BY SODIUM

Sodium Reactor Experiment (SRE)

Power: thermal, 20 Mw; electrical, 6 Mw (net)
Coolant temp.: inlet, 500°F; outlet, 960°F
Fuel element: type, cylindrical; fuel, uranium metal slugs (0.75 in. O.D.); enrichment, 2.8%; clad, stainless steel (0.79 in. O.D.); bond, Na bonded

Experimental Breeder Reactor-II (EBR-II)

Power: thermal, 62.5 Mw; electrical, 20 Mw (net)
Coolant temp.: inlet, 700°F; outlet, 900°F
Fuel element (fuel type No. 1): type, cylindrical; fuel, U–2 wt.% fissium alloy rods (0.144 in. O.D.); enrichment, 49%; clad, stainless steel (0.174 in. O.D.); bond, Na bonded
Fuel element (fuel type No. 2): type, cylindrical; fuel, U–20 wt.% Pu–10 wt.% fissium alloy rods; clad, stainless steel; bond, Na bonded

Enrico Fermi Reactor (APDA)

Power: thermal, 300 Mw; electrical, 100 Mw
Coolant temp.: inlet, 550°F; outlet, 800°F
Fuel element: type, cylindrical; fuel, U–10 wt.% Mo alloy rods (0.148 in. O.D.); enrichment, 28%; clad, zirconium (0.158 in. O.D.); bond, metallurgical

Experimental Breeder Reactor-I (EBR-I NaK Cooled) (Mark III Core)

Power: thermal, 1.4 Mw; electrical, 0.2 Mw
Coolant temp.: inlet, 442°F; outlet, 600°F
Fuel element: type, cylindrical; fuel, uranium metal rods (0.364 in. O.D.); enrichment, highly enriched; clad, stainless steel (0.404 in. O.D.); bond, NaK bonded

Dounreay Plant (DFR)

Power: thermal, 60 Mw; electrical, 15 Mw
Coolant temp.: inlet, 390°F; outlet, 650°F
Fuel element: type, tubular; fuel, U metal tubes (1.00 in. O.D.) (0.40 in. I.D.); enrichment, 40%; clad vanadium (0.020 in. thick (inside of uranium tube); niobium (0.020 in. thick) (outside of uranium tube); bond, Na bonding

Table V-7 GENERAL CHARACTERISTICS OF UO_2 TYPE ELEMENTS IN WATER-COOLED REACTORS

	Fuel Material *
General properties:[1–10] (refer to Table V-1)	
Mechanical[10]	Extremely brittle; fractures under thermal stress.
Melting point[5,6]	High; however, poor thermal conductivity brings the center temperature of fuel pellets under typical operating conditions close to the melting point.
Thermal conductivity	Low; drops to approximately 1 Btu/(hr)(ft)(°F) for reactor operating conditions.
Density	Low — 10.97 g/cm^3 theoretical; the ratio of (metal atoms/cm^3 in UO_2)/(metal atoms/cm^3 in metal) = 51.5%
Thermal stability	Poor; gross radial cracking occurs with some circumferential cracking. No further cracking is noted after the initial.
Radiation stability[1–3,9,10]	Excellent; no indication of any changes other than those due to thermal effects.
Fission-product release:[1–3,11–13]	
Fission gases	40 to 50% of the gases released from 65% dense packed UO_2; approximately 1% released from 95% of theoretical density pressed and sintered pellets.
Other fission products	Small particles of UO_2, which form because of the severe thermal fracturing, can be released to the coolant.
Corrosion[1,3,14]	Excellent resistance in 650°F water and 650 to 750°F steam. Inert in water containing any measurable excess of hydrogen.
Chemical reaction	No exothermic reaction.
Bonding:	
Helium	Elements require a gap between the fuel pellet and the can which changes when fuel pellets fracture. Under irradiation the fission gases mix with the helium, lowering its conductivity and increasing the pressure in the fuel can; low thermal conductivity of gas requires close tolerance of gap to maintain low ΔT. Clad failure allows water and steam to replace helium, leading to further failure.
Lead	Has better thermal conductivity than helium and allows the use of larger gaps between the fuel pellets and fuel can. Little data exist on compatibility of lead with canning material at high temperature.
Temperature limitations[3,11]	Maximum operating temperature of the UO_2 is probably limited to the melting point of 5000°F until more data are available about the failure rate with center melting. Other limitations are associated with the compatibility with the clad materials.
Compatibility with clad materials:	
UO_2-aluminum[15]	No reaction below 400°C.
UO_2-Zircaloy-2[4,16,17]	No reaction below 600°C; slow reaction at 700°C; rapid reaction at 870–1100°C.
UO_2-stainless steel	No reaction below 1200°C.
Fuel lifetime	10,000 Mwd/ton or more expected.
	Canning Material †
Aluminum	Poor mechanical properties at elevated temperatures; low melting temperature; poor corrosion resistance in water[2,3,18,19] metal-water reaction possible;[1,20–24] good thermal conductivity; no reaction with UO_2 below 400°C;[15] low neutron absorption cross section (0.013/cm); low cost.
Aluminum-nickel alloy (M-388)	Poor mechanical properties at elevated temperatures; low melting point; improved corrosion resistance in 500°F distilled water;[25–28] complete oxidation in superheated 840°F (300 psi) steam;[25] good thermal conductivity; metal-water reaction possible;[1,20–24] medium cost; higher absorption cross section (0.0151/cm) than aluminum but still quite low.
Zircaloy-2	Good mechanical properties at 650°F;[2,29–32] high melting point; good corrosion resistance in water at 650°F;[2] relatively low thermal conductivity; metal-water reaction possible;[1,20–24] UO_2-zirconium couples show a diffusion reaction above 1200°F;[4,17] fretting corrosion a problem where vibration exists;[33] hydriding of the clad can occur;[33] low neutron absorption cross section (0.0076/cm); high cost of metal and of close tolerance tubes.
Stainless steel	Excellent mechanical properties at elevated temperatures; high melting point; excellent corrosion resistance in 650°F water; relatively low thermal conductivity; no reaction with UO_2 below 1200°C; metal-water reaction may be possible because of high chromium content;[1,21] high cross section (0.267/cm; stainless steel 304).

*Pressed and sintered UO_2, 92–95 per cent of theoretical density. Applies to all fuel elements in Table V-4 using UO_2 fuel.

†Refer to Table V-3 for general properties.

Table V-8 GENERAL CHARACTERISTICS OF URANIUM METAL TYPE ELEMENTS IN WATER-COOLED REACTORS

	Fuel Material *
General properties (Refer to Table V-2):	
Mechanical	High strength at low temperature; ductility at room temperature decreases with irradiation.
Melting point	Intermediate (2431°F – $\frac{1}{2}$ the melting point of UO_2).
Thermal conductivity	High (~18 times that of UO_2).
Density	High (maximum available for fuel).
Alloying elements	Low-cross-section additives used.
Radiation stability[1-3,34-48,61]	Dimensional stability is good for specimens of this alloy for burn-up of ~0.35% of uranium atoms. In general, metallic fuel with low percentages of alloying elements undergo severe dimensional changes at burn-ups of over 4000 Mwd/ton and at temperatures of <400°C. Above 400°C, swelling type damage limits the burn-up of most high uranium content metal alloys to 3000–4000 Mwd/ton (see Fig. 1). Creep is accelerated under irradiation.
Fission-product release:	
Fission gases	Retained in the metal even at high burn-ups; believed to produce swelling of the uranium.
Other fission products	Retained in metal. In case of a clad failure uranium metal will corrode very rapidly in water at 450°F, releasing the majority of the fission products.
Corrosion[1,2,34-36]	Has poor corrosion resistance in 450°F water but much better than unalloyed uranium; fission products are released by the corrosion.
Chemical reaction	Metal-water reaction is possible.[1,20-22]
Bonding	A metallurgical bond is formed by roll cladding the plate type element.
Temperature limitations	Exceeding the α-β transformation temperature of ~1228°F causes dimensional changes. Swelling damage increases with burn-up and irradiation temperature.[43-46] At 1000°F (% density decrease/% total atom burn-up) ≈ 30 (see Fig. 1).
Compatibility with clad material	Is compatible with Zircaloy-2 below 600°C.[49]
Fuel lifetime	~4000 Mwd/ton expected.
	Canning Material †
Zircaloy-2	Good mechanical properties at 650°F;[2,29-32] high melting point; good corrosion resistance in water at 650°F;[2] poor thermal conductivity; metal-water reaction possible;[1,20-24] UO_2-zirconium couples show a diffusion reaction above 1200°F;[4,17] fretting corrosion a problem where vibration exists;[33] hydriding of the clad can occur;[33] low neutron absorption cross section (0.0076/cm); high cost of metal.

* Uranium alloy (U – 5 wt.% Zr – 1.5 wt.% Nb). Applies to EBWR element only.
† Refer to Table V-3.

Table V-9 GENERAL CHARACTERISTICS OF URANIUM METAL TYPE ELEMENTS IN CO_2-COOLED REACTORS

	Fuel Material *
General properties:[50] †	
Mechanical	High strength at low temperatures; ductility at room temperature decreases with irradiation.
Melting point	Intermediate (2431°F – ½ the melting point of UO_2).
Thermal conductivity	High (~18 times that of UO_2).
Density	High (maximum available for fuel).
Radiation stability	Dimensional stability good at low burn-up;[50] swelling reaches limit of ~10% allowable at ~3000 Mwd/ton at the operating temperature; creep is accelerated under irradiation.[47,48]
Fission-product release:	
Fission gases	Retained in the metal even at high burn-ups; believed to produce swelling of the uranium.
Other fission products	Retained in metal.
Corrosion	Low; $\sim 0.54 \times 10^{-3}$ in. penetration[50–52] per day at ~800°F.
Chemical reaction	No exothermic reaction with CO_2.
Bonding	Helium-filled; reactor gas pressure mechanically bonds can to fuel.[50]
Temperature limitations	Exceeding the α-β transformation temperature of ~1228°F causes dimensional changes. Swelling damage increases with burn-up and irradiation temperature.[43,45,46] At 1000°F (% density decrease/% total atom burn-up) ≈ 30 (see Fig. V-1).
Compatibility	No reaction with magnesium alloy (Magnox A-12) below 450°C.[53]
Fuel lifetime	3000 Mwd/ton expected.
	Canning Material ‡
Magnesium alloy (Magnox A-12)[1,53]	Poor mechanical properties at elevated temperatures; low melting point; good corrosion resistance to CO_2 at 450°C; excellent thermal conductivity, allows efficient finning; very low cross section; low cost; no ignition at 700°C in CO_2 flowing at 30 cu ft/min at atmospheric pressure; ignition occurs in stagnant CO_2 at 640°C and 160 psi.[53]

*Natural uranium.
†Refer to Table V-2.
‡Refer to Table V-3.

number of years in EBR-I and are now in use also in the Sodium Reactor Experiment (SRE). Similar elements, utilizing a uranium-fissium alloy, are to be used in the EBR-II. The Dounreay elements are the same in principle but are annular in shape and use niobium and vanadium as the jacket materials. The element for the Enrico Fermi fast reactor (APDA) is of a different type, consisting of a coextruded zirconium-clad pin of uranium-molybdenum alloy. Its feasibility can possibly be considered proved on the basis of irradiation tests and general experience with the materials.

Involved in the performance of fuel elements of the type mentioned are the performance of the fuel material, that of the jacketing or cladding material, and the combined behavior of the two as assembled into a fuel element. Tables V-1 and V-2 list the general properties of UO_2 and uranium metal, and Table V-3 lists the properties of jacketing materials that have been mentioned. In the case of uranium-metal fuels, the effects of alloying materials must be considered, and consequently Table V-2 cannot be considered to give the properties of the fuel materials for these cases. In Tables V-4 to V-6 are listed the reactors which use, or expect to use, fuel elements of the types mentioned, and the elements are described. In Tables V-7 to V-11, the characteristics of UO_2 and uranium-metal elements are listed in relation to the general reactor types for which they may be employed. In the following paragraphs this information is amplified briefly by discussion. References for details of the data are shown in the tables and are not repeated.

Table V-10 GENERAL CHARACTERISTICS OF UO_2 TYPE ELEMENTS IN GAS-COOLED REACTORS

	Fuel Material *
General properties:[1–10] (Refer to Table V-1)	
Mechanical[10]	Extremely brittle; fractures under thermal stress.
Melting point[5,6]	High; however, poor thermal conductivity brings the center temperature of fuel pellets under typical operating conditions close to the melting point.
Thermal conductivity	Low; drops to approximately 1 Btu/(hr)(ft)(°F) for reactor operating conditions.
Density	Low—10.97 g/cm^3 theoretical; the ratio of (metal atoms/cm^3 in UO_2)/(metal atoms/cm^3 in metal) = 51.5%.
Thermal stability	Poor; gross radial cracking occurs with some circumferential cracking. No further cracking is noted after the initial.
Radiation stability[1–3,9–10]	Excellent; no indication of any changes other than those due to thermal effects.
Fission-product release:[1–3,11–13]	
Fission gases	40 to 50% of the gases is released from 65% dense packed UO_2; approximately 1% is released from 95% of theoretical density pressed and sintered pellets.
Other fission products	Nonvolatiles retained in UO_2.
Corrosion:[1,3,14]	Essentially inert in CO_2 gas.
Chemical reaction	No exothermic reaction with CO_2.
Bonding (helium)	Elements require a gap between the fuel pellet and the can which changes when fuel pellets fracture. Helium bonding is used as the heat-transfer medium. Under irradiation the fission gases mix with the helium, lowering its conductivity and increasing the pressure in the fuel can; low thermal conductivity of gas requires close tolerance of gap to maintain low ΔT.[10]
Temperature limitations[3–11]	Maximum operating temperature of the UO_2 is probably limited to below the melting point of 5000°F until more data are available about the failure rate with center melting. Other limitations are associated with the compatibility with the clad materials.
Compatibility with clad materials (stainless steel)	No reaction below 1200°C.
	Canning Material
Stainless steel	Excellent mechanical properties at elevated temperature; high melting point; excellent corrosion resistance to CO_2 for temperatures <1400°F; relatively poor thermal conductivity, ineffective finning; no reaction with UO_2 below 1200°C; high cross section (0.267/cm; stainless steel 304).

*Pressed and sintered UO_2, 92–95 per cent of theoretical density.

Table V-11 GENERAL CHARACTERISTICS OF URANIUM METAL ALLOY ELEMENTS IN SODIUM-COOLED REACTORS

Fuel Material (Uranium Alloys)	
General properties: (Refer to Table V-1)	
Mechanical	High strength at low temperature; ductility at room temperature decreases with irradiation.
Melting point	Intermediate (2431°F – ½ the melting point of UO_2).
Thermal conductivity	High (~18 times that of UO_2).
Density	High (maximum available for fuel).
Alloying elements:	
Molybdenum	High thermal neutron absorption cross section.
Fissium	High thermal neutron absorption cross section.
Fissium and molybdenum	Low neutron absorption cross section for fast reactors.
Radiation stability:[1-3,34-48,61]	
Unalloyed uranium (See Table V-8).	
U–20 wt.% Pu–10.8 wt.% Fs alloy specimens	Show good dimensional stability at ~1000°F and 0.5 total at.% burn-up. Elongation amounted to 1.8% per at.% burn-up.[54-56]
U–5 wt.% Fs alloy specimens	Show good dimensional stability at central temperatures in the range 700 to 1100°F at a burn-up of 0.76% of all atoms.[57]
U–10 wt.% Mo alloy specimens	Show good dimensional stability at central temperatures of ~1000°F, interface temperature of 400°F, and burn-up as high as 10,000–20,000 Mwd/ton max.[58]
Fission-product release:	
Fission gases	Retained in the metal even at high burn-ups; believed to produce swelling of the uranium.
Other fission products	Retained in metal. Low corrosion with clad failure.
Corrosion[1,2,34-36]	Excellent corrosion resistance in Na or NaK up to 600°C.
Chemical reaction[1,20-22]	None with sodium; U–20 wt.% Pu–10 wt.% Fs alloy is pyrophoric in swelled condition.[56,59,60]
Bonding	Sodium or NaK provides a good thermal bond.
Temperature limitations[43-46]	Exceeding the α–β transformation temperature of ~1228°F causes dimensional changes; swelling damage as a function of temperature appears to be less for these alloys.
Compatibility with clad material:[49]	
Zircaloy-2[2,29-33]	Compatible below 600°C.
Stainless steel	Uranium forms low melting eutectics with iron, nickel, and chromium which melt at temperatures of 750 to 850°C; diffusion zones formed at ~500°C.[70]
Niobium	Compatible below 1000°C; fails in ~1 hr in contact with uranium at 1300°C.[71]
Vanadium	Compatible below 700°C; fails radially at ~1000–1100°C, forming a low-melting eutectic with uranium having a melting point of 1130°C.[71]
Fuel lifetime	8000 Mwd/ton or more expected for the fissium and molybdenum alloys.
Canning Material (Refer to Table V-3)	
Zircaloy-2	Good mechanical properties;[2,29-32] high melting point; good corrosion resistance in sodium but oxygen impurities tend to oxidize zirconium;[72] relatively poor thermal conductivity; compatible with uranium below 1200°F;[49] low neutron absorption cross section (0.0076/cm); high cost of metal.
Stainless steel	Excellent mechanical properties at elevated temperatures; high melting point; corrosion resistance in sodium; relatively poor thermal conductivity; uranium forms low melting eutectics with iron, nickel, and chromium, which melt at temperatures of 750 to 850°C; diffusion zones formed at ~500°C;[70] high cross section (0.267/cm); S.S. 304.
Niobium[69,71]	Good mechanical properties at >1000°F; high melting point; compatible with sodium; thermal conductivity better than stainless steel; relatively high neutron absorption cross section (0.059/cm); reacts with uranium rapidly at 1300°C.
Vanadium[69,71]	Good mechanical properties at elevated temperature; high melting point; compatible with sodium; relatively poor thermal conductivity; high neutron absorption cross section (0.306/cm); reacts with uranium rapidly at between 1000–1100°C.

Oxide Elements in Water-cooled Reactors

The outstanding advantages of oxide elements (UO_2 or ThO_2) in water-cooled reactors are their chemical compatibility with water and their excellent resistance to radiation damage. These two properties are expected to make possible the long fuel lifetimes which are necessary for economic operation with partially enriched fuels. The good radiation-damage resistance should allow a high burn-up before the element becomes unusable. The resistance to aqueous corrosion should minimize the effects of jacket failures when they do occur and should allow the elements to be used in the reactor for a residence time which approaches the corrosion lifetime of the jacket material, if such long residence proves necessary.

The major limitation imposed on reactor performance by these elements arises from limitations on heat conduction from the point of heat generation to the jacket material. Thermal resistances originate in the poor thermal conductivity of the oxides and in the lack of a metallurgical bond between the oxide material and the jacket. No metallic bond has been proved for this use; however a lead bond has been used experimentally in the BORAX-IV reactor. Large power reactor designs currently depend on a helium thermal bond in conjunction with small clearances between the two materials.

The current cost of fabricating complete assemblies of these elements is rather high. The high costs result from the necessity of building the assemblies from units that are fundamentally small and from the necessity for maintaining close tolerances in order to minimize the gap between the oxide pellets and the jackets in the assembled elements. The development of better schemes for thermal bonding would reduce the latter sources of cost, whereas the former should be reduced if the production volume becomes large enough to make extensive mechanization feasible.

There are a number of areas in which future experimental information is needed. The maximum permissible operating temperature of the oxide is not well established. It is conceivable that the material could be operated at temperatures up to its melting point or that some melting could be tolerated. It is known that the oxide elements crack in operation, evidently from the effects of thermal stress. The long-term effects of such cracking cannot be completely predicted from experiments on a limited number of samples. The effects of fission-gas release need further study. The rate of release depends on the density of the oxide and the operating temperature and is small for high-density materials if the melting point is not approached. The effects must be considered if operation at very high temperature is contemplated. The effect of the fission gases in reducing the thermal conductivity in the gap between the oxide and the jacket is important, as is also the pressure build-up in the jacket.

Both Zircaloy and stainless steel are used or will be used for jacketing oxide elements. Both these materials are probably adequate in corrosion resistance for any temperatures currently contemplated for water-cooled reactors, although there is some question of the performance of zirconium if the residence time of the fuel must be extremely long (several years), and there may be problems from the activities of corrosion products that come from steel jackets. The major considerations, however, in choosing between the two jacket materials are those of cost and possible metal-water reactions. The cost consideration involves a compromise between the somewhat lower cost of the material and its fabrication in the case of stainless steel and the higher fuel costs that result from the poorer neutron economy. Molten zirconium is known to react violently with water under certain circumstances. These circumstances are not expected in power reactors, but they can be hypothesized for certain accidents. A stainless-steel–water reaction represents, potentially, a much smaller energy release, and the reaction has not been demonstrated. A reaction has been observed with Nichrome-V in water and must be attributed to the chromium content of the Nichrome-V. The chromium content of stainless steel is little different.

Metallic Fuel Elements in Water-cooled Reactors

It would be desirable to use metallic fuel elements with metallurgically bonded jackets in water-cooled reactors because the higher density of uranium attainable in such elements reduces the relative absorption by parasitic absorbers, because the high thermal conductivity

of the elements make possible higher power densities, and possibly because the elements may be cheaper to fabricate. Despite these advantages the metallic element has been abandoned in current H_2O-cooled designs because of its poor resistance to aqueous corrosion and its relatively poor resistance to radiation damage. In addition, it is generally felt that such elements are more susceptible to the possibility of violent metal-water reactions than are the oxide elements. The corrosion and radiation-damage considerations are strongly influenced by the necessity for long fuel lifetime if enriched uranium is used as fuel. In natural-uranium aqueous reactors, these considerations are somewhat less important, and metal fuel elements are considered (pp. 59-63).

In the early development of water-cooled reactors, metallic fuels were investigated quite intensively. It was found possible to make improvements in corrosion resistance and radiation damage individually by suitable alloying and heat-treating procedures, but no fuel material was developed which combined good corrosion resistance and good radiation-damage resistance. If breaks occur in the cladding of metallic fuel elements, it is expected that the corrosion of the fuel will be quite rapid. Rather severe contamination of the water coolant will probably occur, and it is possible that the distortions of the fuel element caused by the corrosion may block adjacent cooling channels and lead to the progressive failure of other elements.

The EBWR fuel elements are metallic plates clad with Zircaloy-2. The fuel is an alloy of uranium, 5 per cent zirconium, and 1.5 per cent niobium. The niobium addition was primarily to improve the fabrication properties of the plates. These elements have been in operation in the EBWR for approximately $1\frac{1}{2}$ years, and no failures have been reported. The fuel burn-up has not been high because much of the operation has been at low power densities and under experimental conditions. The average burn-up at this date is probably somewhat less than 1000 Mwd/ton; the maximum burn-up is perhaps three times the average value.

Practically all the past development on metallic fuel elements for water reactors has involved the use of zirconium as the clad material. It bonds metallurgically to uranium and constitutes quite an acceptable clad. Stainless steel forms low melting eutectics with uranium, and there seems to be little incentive for considering it.

Highly Enriched Fuel Elements for Water-cooled Reactors

Highly enriched fuel elements do not have wide fields of application for central-station power reactors. They may be useful for ship-propulsion reactors and may find some application in central-station reactors as seed regions or spikes. The Zircaloy-clad zirconium-uranium alloy plates used in the Shippingport reactor have been proved through rather long use and are highly satisfactory in their performance. Their lifetime is probably more than adequate, since the burnout of fuel from a fully enriched element will place other limits on lifetime. The steel elements used in the APPR have given satisfactory performance up to this point and may also be considered as a proven type. These elements are discussed further in references 77 and 78.

Metallic Fuel Elements for Gas-cooled Reactors

The gas-cooled reactors that have been built to date are natural-uranium reactors. In such reactors the great advantage of uranium-metal fuel elements is the high density of uranium which they provide and the relatively good thermal conductivity, which allows the use of rather massive elements. Both these characteristics tend to minimize the absorption by parasitic materials and the resonance absorption of U^{238}. Consequently, natural-uranium reactors fueled with metallic elements characteristically exhibit substantially higher reactivity than would be attainable in similar oxide-fueled reactors. A further loss of reactivity in the oxide-fuel reactor results from the relatively large effect of Doppler broadening of absorption resonances in U^{238} because of the higher operating temperature of the fuel.

The important limitations of the metallic fuel elements in gas-cooled reactors are their temperature limitations and their susceptibility to radiation damage. These two characteristics are interrelated. The important result of radiation

damage appears to be the swelling of the uranium, presumably caused by fission-product gases, after the irradiation has reached a few thousand megawatt days per ton. The degree of swelling is also a function of the uranium temperature. Results of measurements on this effect are shown in Fig. 1. The second temperature

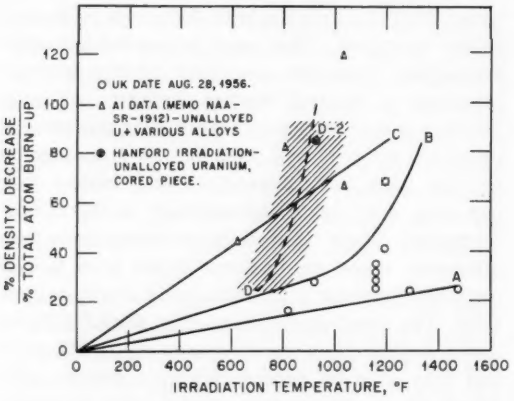

Fig. 1—Volume change vs. irradiation temperature. Unalloyed uranium plus various alloys. A, lower limit of unalloyed uranium data, UK data, Aug. 28, 1956. (Unpublished.) B, approximates best fit to unalloyed uranium, UK data, Aug. 28, 1956. (Unpublished.) C, lies above ~80 per cent UK data for uranium and uranium alloys, M. Finniston, UKAEA, Nov. 1, 1957, private communication. ▨, range of temperature where US and UK information indicates a nonlinear decrease in density. The bulk of US data lies within this region.

limitation on Calder Hall type metal fuel elements is the limit of the magnesium alloy fuel jacket. It appears that the coolant temperatures that can be attained with such fuel elements will be limited to about 800°F. Although the jacket temperature is the more immediate limitation, it appears that the swelling of the uranium will set a limit in the same range, unless ways can be found of improving this characteristic. If this can be done and if a jacket of higher temperature capability, such as beryllium, can be developed, the metal element may continue to be of great interest for gas-cooled reactors.

The Calder Hall type elements have been in operation for a relatively long time, and they appear to be satisfactory, provided their temperature limitations are not exceeded and irradiation to high burn-up is not required.

Oxide Elements for Gas-cooled Reactors

The uranium oxide fuel element has not been used in a gas-cooled reactor, but it would seem reasonable to presume that elements of this type, jacketed with stainless steel, would be quite feasible for temperatures within the permissible limits for stainless steel. Operation with natural uranium is, of course, not possible with such elements. The other limitations on performance will be of the same type as those which apply for uranium oxide in water-cooled reactors, but the consequences of low thermal conductivity will be less severe because lower heat fluxes will be used. Indeed, the high surface temperature that can be used with stainless-steel jackets will improve quite markedly the thermal performance of gas-cooled reactors with respect to power density as well as with respect to maximum coolant temperature.

Metallic Elements for Sodium-cooled Reactors

Sodium-cooled reactors include both fast reactors and thermal or near-thermal reactors (sodium-graphite). Uranium metal is resistant to corrosion by sodium, and consequently one of the objections to its use in aqueous reactors is eliminated. In fast-neutron reactors rather large proportions of alloying material of cross section too high to be tolerable in thermal reactors may be used. Consequently, the radiation-damage resistance of the metal can be greatly improved by appropriate alloying, notably with fissium and with molybdenum. Because of the high heat-transfer coefficients which are characteristic of sodium, the steam temperature that can be attained in the nuclear power plant is little different from the maximum fuel-element surface temperature; consequently, the temperature limitations of the fuel-element metal are not so important when this coolant is used. Because of these circumstances, it would appear that the metallic fuel element has a promising future in the fast reactor. The situation is not so clear in the sodium-graphite reactor, and it may be that oxide elements will come into use because of their greater resistance to radiation damage

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RADIATION DAMAGE TO OTHER REACTOR MATERIALS

The materials used in the construction of nuclear reactors and certain associated equipment are subjected in use to high intensities of nuclear radiations. A brief review is given of the data that are available on the effects of radiation damage to materials that may be used in or around the reactor. The discussion is limited to materials other than fuel materials.

The radiation originating in the fission process, either immediately or with some time delay, consists of fission fragments, fast neutrons, gamma radiation, and beta particles. By interaction with the reactor materials the neutrons may be slowed down to thermal energy, and, consequently, thermal neutrons also become part of the reactor radiation. The thermal neutrons do not ordinarily produce any damage by transferring energy to the irradiated material; they can, however, react with the nuclei of the material, giving rise to secondary radiations — gamma rays, alpha particles, protons, and beta particles. These secondary radiations may in turn produce damaging effects. The mechanisms of radiation damage by reactor radiation are summarized briefly in Table VI-1.

By far the most damaging of the radiations listed are the fission fragments. Their energy is very high, and their range is short; consequently, each fission fragment produces a small region of concentrated damage. In solid-fuel reactors the damage is confined to the fuel elements and possibly to bonding or jacketing material that is in contact with the elements.

Besides fuel materials, the types of materials which are ordinarily of interest for reactor applications are metals, ceramics, organic materials, and certain simple inorganic compounds such as H_2O. Among these classes of materials the susceptibility to radiation damage varies widely. In the metals and most of the ceramics of interest, damage is caused by the displacement of atoms in the crystal structure. Most of these materials are highly resistant to damage and are only slightly affected by radiations other than fast neutrons or charged particles. Organic materials are damaged by the breaking of the chemical bonds of the molecules. They are very much more susceptible to radiation and are damaged significantly by all types of nuclear radiation. Compounds such as H_2O are damaged by the same process, but, in the case of simple molecules, recombination can occur to yield the original compound, and the net damage may be small.

For the materials that are damaged primarily by fast neutrons, it is customary to express the radiation dose as the time integral of the neutron flux (nvt), specifying whether the neutrons are fast or thermal and, if possible, giving some description of the fast-neutron spectrum. For the materials that are damaged by all types of radiation, other units must be used. Common units are the Roentgen, r, and the rep. These units are based on the amount of energy which would be deposited by the radiation in a unit volume of a standard material. The use of such units is based on the observation that the radiation damage produced by a given type of radiation is usually proportional to the amount of

Table VI-1 MECHANISM OF RADIATION DAMAGE BY PILE RADIATION[1]

Damage by fast neutrons:
Scattering (recoil ion; proton ejection)
Displacement (neutralized recoil ion)

Damage by gamma rays:

	Atomic No.	
	27 or less	82 or greater
Photoelectric	<0.1 Mev	<0.5 Mev
Compton effect	0.1–10 Mev	1.0–4 Mev
Pair production	>10 Mev	>4 Mev

Damage by thermal neutrons:

Neutron-gamma reaction
$H(n,\gamma)D$
$Cl^{37}(n,\gamma)Cl^{38}$

Neutron-proton reaction
$N^{14}(n,p)C^{14}$

Neutron-alpha reaction
$B^{10}(n,\alpha)Li^{7}$
$Li^{6}(n,\alpha)H^{3}$

Fission-fragment damage: Direct ionization;
Displacement (neutralized fission ion)

Beta-particle damage: Direct ionization

energy deposited in the material. A more recently devised unit is the rad. Any material is said to have experienced 1 rad of radiation when it has absorbed 100 ergs of energy from the radiation per gram of material. Definitions of the various dose units are summarized as follows:

r (Roentgen), the quantity of X or gamma radiation such that the associated corpuscular emission will produce one e.s.u. of charge (corresponding to 2.08×10^9 ion pairs) of either sign in 1 cm^3 of air at standard conditions. This corresponds to the absorption of 83 ergs per gram of air.

rep (Roentgen equivalent physical), that quantity of ionizing radiation which will result in the absorption in animal tissue of 93 ergs per gram.

rad, that radiation dose which corresponds to the absorption of 100 ergs per gram in the material under consideration.

rem, that quantity of radiation which, when absorbed in living tissue, produces a biological effect equivalent to the absorption of 1 rad of X or gamma rays.

Figure 2 shows roughly the ranges of doses which can be tolerated by materials of several classes before serious radiation damage occurs.

Table VI-2 AMOUNT OF ENERGY ABSORBED BY COMMON MATERIALS FROM THE SAME RADIATION FIELD[2]*

Material and proportion elements	Energy absorbed per hour (ergs/g) $\times 10^{-8}$
Polyethylene 1C : 2H	2.4
Natural rubber 1C : 1.6H	2.0
Polystyrene 1C : 1H	1.6
Lucite (polymethylmethacrylate) 1C : 1.6H : 0.4O	1.6
Nylon 1C : 1.83H : 0.16O : 0.16N	2.2
Tissue 1C : 8H : 3.6O : 0.2N†	1.96

*The radiation field inside the ORNL Graphite Reactor consists of the following approximate fluxes:
1.1×10^{12} thermal neutrons/(cm^2)(sec)
1.4×10^{11} neutrons (≥ 0.1 Mev)/(cm^2)(sec)
6.7×10^{10} neutrons (≥ 0.5 Mev)/(cm^2)(sec)
4.2×10^{10} neutrons (≥ 1.0 Mev)/(cm^2)(sec)
$\sim 5 \times 10^{11}$ gamma protons (av. E = 1 Mev)/(cm^2)(sec)

†Based on an approximate composition for tissue of $(C_5H_{40}O_{18}N)$.[2]

In general, the damage to a particular material increases with increasing radiation dose, and there is little effect of dose rate except in electrical insulators.

Evidently the rate of absorption of radiation of a given type and intensity will vary with materials, and the relative absorption of the different components in a mixed radiation field will vary also with the material. The absorption of gamma energy per unit mass of material will depend on the atomic number of the atoms in the material, whereas the rate of neutron-energy absorption will depend primarily on the percentage of light elements, particularly hydrogen, in the material. Since the percentage compositions of organic materials do not vary over a wide range, the absorption characteristics of these materials do not vary greatly. There is some variation, which is illustrated in Table VI-2, which shows the energy absorption by

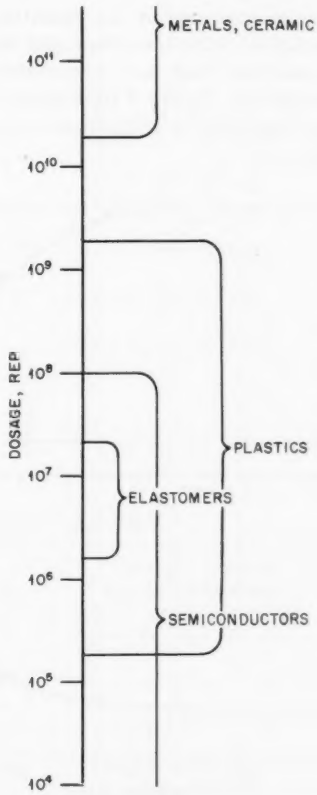

Fig. 2—Relative stability of materials to physical property changes by radiation fields composed of 10 rep of gamma photons to 1 rep of neutrons. (Data taken from reference 3.)

various organics in a field of mixed radiation.

Data on damage to elastomers, plastics, and organic liquids, as well as on components using such materials, are found in references 1, 3, and 4. Representative data from these reports are given in Figs. 3 to 6 and in Tables VI-3, VI-4, and VI-5. The radiation mixture was approximately the same for the irradiations summarized in the three tables and is specified in the footnote to Table VI-3.

One of the most frequently used components for instrumentation is the thermocouple. Thermocouples are not listed in Table VI-4 since they are metallic and are very radiation-resistant. However, reactor radiation will generate heat in the thermocouple and can lead to a false indication of the temperature in cases where the generated heat is significant (e.g., in measurements of gas temperatures).

The service life under irradiation of many instruments, electrical components, and electrical wire is governed by the radiation damage to the electrical-wire insulation and other electrical insulators that are composed of solid organic material. Figure 5 illustrates the damage to be expected in components using these materials.

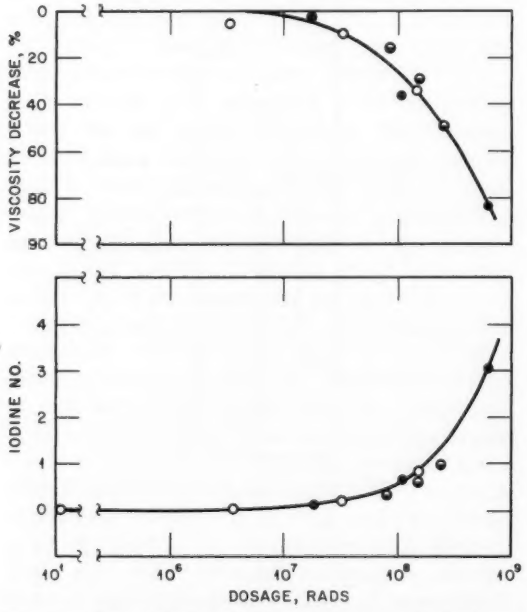

Fig. 3—Comparison of pile, gamma, and electron radiation damage to glycerine. ●, gamma radiation. ○, pile radiation. ◒, electron radiation. (Data taken from reference 2.)

Electrical insulators are a class of materials which are affected by radiation dose rate. The damage is one of change in the conductivity of the material and may lead to changes in operational characteristics of the component in which the material is used. Ionization in the material

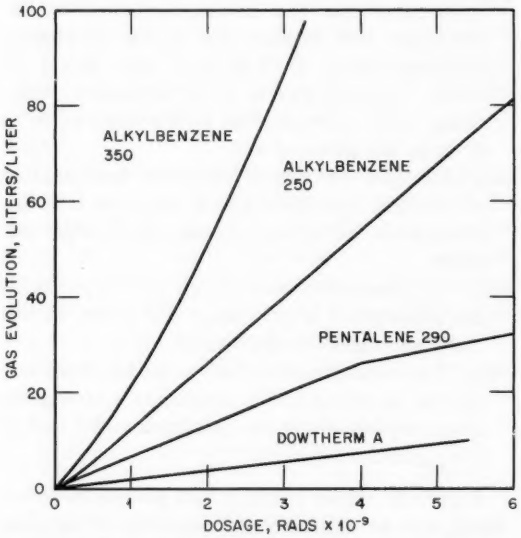

Fig. 4—Gas evolution from alkylbenzene MW 350, alkylbenzene MW 250, Pentalene 290, and Dowtherm A as a function of radiation dosage. In contrast to viscosity changes, gas evolution is an approximately linear function of dosage. (Data taken from reference 2.)

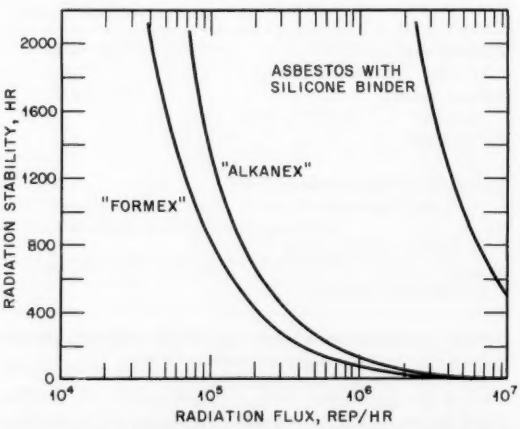

Fig. 5—Radiation stability of electrical components that use asbestos with a silicone binder and Alkanex and Formex as insulating materials. (Data taken from reference 3.)

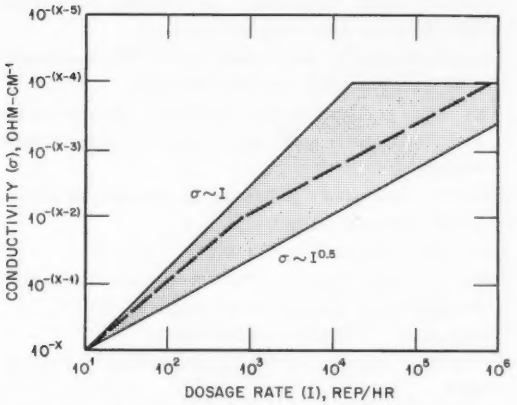

Fig. 6—Radiation-induced conductivity in electrical insulators. ——, theoretical upper and lower limits of the induced conductivity. ---, general shape of the expected curve of conductivity in typical materials. (Data taken from reference 3.)

Table VI-3 RADIATION DAMAGE TO REACTOR AND SHIELD COMPONENTS[1]

Moderator, coolant, and shield solutions	Radiation damage	Est. damage, rads*†
Water	Gas evolution, corrosion	Infinite
Paraffin and olefin-hydrocarbons	Gas evolution, polymerization	1×10^8
Monophenyl hydrocarbons	Gas evolution, polymerization	3×10^8
Di- and polyphenyl hydrocarbons	Gas evolution, polymerization	1.4×10^9

*25 per cent damage.

†1×10^8 rads = 1×10^{10} ergs per gram of material or 1000 watt-sec per gram or 430 Btu per pound. Corresponds approximately to combined radiation of 3×10^{17} thermal neutrons/cm^2, 3×10^{16} fast neutrons/cm^2 (above 0.7 Mev), and 1.5×10^{17} gammas/cm^2 (average energy = 1 Mev).

Table VI-4 RADIATION DAMAGE TO CONTROLS AND INSTRUMENTATION COMPONENTS[1]

Components	Radiation susceptible material	Estimated threshold damage (25% damage)
Capacitors	Organics Paper or oil dielectric; bakelite or waxed-paper cover	$(4.5 \text{ to } 7.0) \times 10^6$ rads
Resistors	Plastics Bakelite cover for resistance material	$(4.5 \text{ to } 7.0) \times 10^6$ rads
Microswitches	Plastics Plastic case and actuator	$(4.5 \text{ to } 7.0) \times 10^6$ rads
Selsyns	Electrical insulators (organics) Wire insulation, brush holder, slot liners, end punchings, and casing insulation	$(4.5 \text{ to } 7.0) \times 10^6$ rads

Table VI-5 RADIATION DAMAGE TO HEAT ENGINE COMPONENTS[1]

Components	Radiation susceptible material	Estimated threshold damage (25% damage)
Gaskets and seals	Elastomers Leather	$(4.5 \text{ to } 7.0) \times 10^6$ rads
Hoses and couplings	Rubber and plastic hoses Plastic and fabric couplings	$(4.5 \text{ to } 7.0) \times 10^6$ rads
Fuels, lubricants, and hydraulic fluid	Petroleum base (paraffinic hydrocarbons)	$(1.0 \text{ to } 2.5) \times 10^8$ rads
Antioxidant additives		Threefold improvement noted with some additives
Electric motors and wiring	Electrical insulators (organics)	$(4.5 \text{ to } 7.0) \times 10^6$ rads

which is a function of the rate of radiation is the prime mechanism for this damage. Figure 6 illustrates the change in conductivity that occurs with radiation dose rate for a group of insulators.

A comparison of the data given in Fig. 3 indicates that the damage per unit of energy absorbed in glycerine is approximately the same for gamma, reactor, or electron radiation up to a total dosage of $\sim 10^9$ rads.[3] This is apparently true for a large number of the organic materials up to total dosages of $\sim 10^8$ rads.

Table VI-6 EFFECT OF NEUTRON IRRADIATION ON THE HARDNESS OF SOME METALS[5,6]

Integrated neutron flux (F),* 10^{19} nvt	Hardness,† Bhn	
	Unirradiated	Change
Aluminum 1100		
240	38	+34
720	38	+42
2600	38	+38
Aluminum 356		
240	67	+29
720	67	+34
2600	67	+13
Beryllium QMV		
240	127	+52
720	127	+97
2600	127	+52

*(F) refers to fast neutrons. The irradiation temperature was not given.
†Converted from Rockwell hardness values.

In the case of metallic materials the radiation damage manifests itself as a change in the physical properties of the materials and apparently does not limit the life of the metallic material. The changes are normally not significant except for radiation dosages involving fast neutrons in integrated flux levels above 10^{19} nvt.

Where dynamic stresses, in particular, are involved, the radiation-induced changes, such as the decrease of ductility, may be serious. The general effects of fast-neutron radiation on structural metals are summarized in the following listing[5] (includes carbon steels, stainless steels, aluminum alloys, nickel alloys, cobalt alloys, magnesium alloys, titanium, zirconium, molybdenum, tungsten, tantalum, copper, and beryllium):

Yield strength: Increased up to 450%, annealed, and less for cold-worked metals
Tensile strength: Increased up to 75%, annealed, and less for cold-worked metals
Ductility: Decreased one-fifth to one-third (average), annealed, and less for cold-worked metals
Elastic constants: Little or no change, according to limited data
Work hardening: Decreased
Impact properties: Ductile-brittle transition temperature raised by as much as 100°C
Creep rate: Usually unaffected
Fatigue strength: No effect, according to limited data
Hardness: Increased, generally less than 100 Bhn
Damping capacity and internal friction: No change, according to limited data
Density: Generally decreased by as much as 0.2%
Diffusion coefficient: Slightly increased, according to limited data
Electrical resistivity: Increased, generally by less than 10% at room temperature
Phase changes: Possible in certain systems
Microstructure: No change
Thermoelectric electromotive force: Little change
Thermal conductivity: No change, according to limited data
Corrosion resistance: Minor changes
Residual radiation: Cobalt-containing alloys are undesirable

It is fortunate that the damaging effects of radiation on metals are small because metals must be relied upon as parts of the reactor structure for many years. It is difficult to obtain sufficiently large radiation doses to test metals over ranges corresponding to the desired lifetimes. Some data do exist, however, on aluminum, beryllium, and stainless steel for irradiations up to 2600×10^{19} nvt. Such irradiations would correspond to approximately nine years in a fast neutron flux of 10^{14} neutrons/(cm^2)(sec). These results are summarized in Tables VI-6 and VI-7. It is reassuring to observe that the damage effects do not increase appreciably between 240 and 2600×10^{19} nvt.

This summary of radiation-damage effects is not meant to be a comprehensive coverage of the data. For detailed information on radiation-damage effects, the reader is referred to the

Table VI-7 EFFECT OF NEUTRON IRRADIATION ON THE TENSILE PROPERTIES OF SOME METALS[5,6]

Integrated neutron flux (F),* 10^{19} nvt	Yield strength, 1000 psi		Tensile strength, 1000 psi		Elongation, %	
	Unirradiated value	Change	Unirradiated value	Change	Unirradiated value	Change
			1100-H14 aluminum			
10	16.6	+7.4	17.3	+8.7	6.0	−0.5
			1100 aluminum			
240	18.4	+4.8	20.3	+6.7	22.3	+3.2
720	18.4	+5.2	20.3	+8.9	22.3	−2.1
2600	18.4	+8.1	20.3	+10.1	22.3	+0.7
			356 aluminum			
240	24.1	+18.3	32.4	+11.9	2.7	−2.3
720	24.1	+26.9	32.4	+21.6	2.7	−1.8
2600	24.1	+11.9	32.4	+12.6	2.7	−1.2
			347 stainless steel			
240	61.6	+36.4	95.8	+13.2	63.2	−11.6
720	61.6	+35.7	95.8	+13.9	63.2	−12.6
2600	61.6	+39.2	95.8	+11.4	63.2	−19.2
			QMV beryllium			
240	24.5		35.7	+15.7	1.4	−1.2
720	24.5		35.7	+14.7	1.4	−1.2

*(F) refers to fast neutrons. The irradiation temperature was not given.

Radiation Effects Information Center (REIC) which has been established at Battelle Memorial Institute by the U. S. Air Force to provide a means of collecting and disseminating radiation effects data to research and development organizations.[8–10] Monthly abstracts[7] are prepared of current reports of radiation effects on materials and components including: electrical and electronic systems, polymeric materials, ceramic materials, metals and alloys, inorganic chemicals, organic chemicals, facilities, dosimetry, and experimental techniques.

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PROGRESS ON SPECIFIC REACTOR TYPES

ORGANIC MODERATED REACTOR EXPERIMENT

The use of an organic material as the coolant, moderator, or both has been considered by many reactor engineers as having distinct advantages over other materials. In general, these advantages are (1) high-temperature capabilities at low pressures because of their low vapor pressures, (2) compatibility with fuel, canning, and structural materials resulting in low corrosion rates, (3) low induced radioactivity, and (4) no hazardous chemical reactions.

A reactor using such an organic could therefore be expected to have a relatively low capital cost since carbon or low alloy steels and standard pumps and valves could be used, and reasonably high temperatures could be achieved in a low-pressure system.

However, one of the major unknowns has been the effect of radiation and high temperature on the properties of the organics. Thus much research and development effort has been expended on the selection of an organic that would be stable in a reactor environment. Much of the preliminary work in this field is summarized in references 1 and 2 on organic moderator and coolant technology. Since the early experiments indicated that the aromatic materials, in particular, the polyphenyls, exhibited fair stability, the major effort has been placed on the study of polyphenyls. Recognizing that the feasibility of the organic-moderated reactor concept was dependent on the behavior of the organic under reactor operating conditions, the U. S. Atomic Energy Commission contracted with Atomics International, a division of North American Aviation, Inc., to construct an organic-cooled and -moderated reactor. The contract resulted in the construction of the Organic Moderated Reactor Experiment (OMRE), a pressurized organic-cooled and -moderated reactor having a maximum thermal power output of 16,000 kw. This reactor was specifically designed as a test facility for organics, and no provision was made for the generation of electricity. Detailed information on the design and construction of the reactor and other information on the research and development of organic moderator and coolants have been reported in references 3 to 29.

Recently, Atomics International[30,31] reported some of the results of their experiments with the OMRE, which went critical on Sept. 17, 1957. The general characteristics of the OMRE are listed in Table VII-1.

A summary and discussion of the significant results of the limited operation of the OMRE presented in references 30 and 31 follows.

OMRE Core Characteristics

Table VII-2 lists the core characteristics of this reactor.

From this data it can be seen that this particular core design has a positive temperature coefficient below 450°F and a negative coefficient above 480°F. A temperature-coefficient test run of reactivity[30] on the OMRE showed that this coefficient varied.

Temp., °F	$(\Delta k/k) \times 10^5$, °F
240	+10
450–480	0
700	−10

Thus at the proposed operating temperature (>700°F) this reactor does have a relatively large negative temperature coefficient which represents a stabilizing function to the reactor since approximately 1 per cent reactivity can be held by a temperature change from ~480 to 700°F. However, for a moderator temperature

Table VII-I GENERAL CHARACTERISTICS OF THE OMRE*

General
- Type: Organic moderated
- Location: National Reactor Testing Station, Idaho
- Contractor: Atomics International

Power
- Thermal: 5,000–16,000 kw
- Electrical: None
- Max. heat flux: 389,000 Btu/(hr)(sq ft)
- Av. heat flux: 100,700 Btu/(hr)(sq ft)
- Av. specific power: 786 kw/kg

Fuel and materials
- Fuel material: UO_2
- Fertile material: None
- Amount of fuel (U^{235}): 25.5 kg
- Geometry of fuel: Plates
  - Dimensions: 0.02 in. thick; 2.5 in. wide; 36 in. long; with 0.134 in. space between plates
- Thermal shielding
  - Material: Carbon steel
  - Arrangement: Cylindrical
  - Coolant: Santowax O-M†
- Biological shielding
  - Material: Earth and concrete
  - Coolant: None
- Reactor vessel material: Low alloy steel
- Load and unload fuel system: Semiremote extension tools

Heat removal and operating data
- Primary loops: 1
- Coolant: Santowax O-M†
- Operating pressure: 200 psi
- Inlet temperature: 500 to 700°F
- Outlet temperature: 530 to 710°F
- Maximum cooling velocity in core: 15 fps
- Flow rate: 9200 gpm
- System pressure drop: 55 psi
- Type of circulation: Forced
- Pumping power: 300 hp
- Pumps: 2
- Pump type: Centrifugal
- Heat-exchanger type: Air blast finned tube

Physics data
- Thermal neutron-flux density
  - Max.: 1×10^{14} neutrons/(cm^2)(sec)
  - Av.: 3×10^{13}
- Fast neutron-flux density
  - Max.: 5×10^{14}
  - Av.: 2×10^{14}

Physics data (Continued)
- Fuel element lattice spacing: 4.5 in., center to center
- Plates per assembly: 16
- Fuel cladding material: 304 stainless steel
  - Cladding thickness: 0.005 in.
- Max. sheath temperature: 800°F
- Max. fuel temperature: 824°F
- Metal-to-moderator ratio (atomic): 1 to 10.4
- Expected life of elements
  - Exposure: variable (~2000 Mwd)
  - Per cent burn-up: 11.2

Moderator
- Material: Santowax O-M†
- Amount: 2.72×10^5 cm^3
- Operating temperature: 500 to 700°F

Reflector
- Material: Santowax O-M†
- Amount: 1.12×10^6 cm^3
- Operating temperature: 500 to 700°F

Reactivity balance
- Leakage: 0.25
- Flattening: None
- Fuel depletion: 7.15%
- Temperature coeff.: Variable
- Xenon and poisoning: 3.54%
- Total reactivity to control: 10.7%
- Average neutron energy: 0.045 ev
- Neutron lifetime: 120 μsec

Principal dimensions
- Core: 22.5 in. square by 36 in. high
- Pressure vessel: 4.5 ft in diameter by 24 ft high
- Reflector: 8 in. thick
- Thermal shield: 2 in. thick
- Biological shield: 14 ft thick

Controls and containment
- Rods: 12
- Shape: Cylindrical
- Poison: Boron carbide
- Drive mechanism location: Outside, above reactor vessel
- Type of drive: Rack and pinion
- Operating control method: Auto flux control
- Scram control method: De-energize electromagnetic latch
- Other safety features: Auxiliary coolant loop
- Containment: None
- Maximum credible accident: Release of core fission products
- Exclusion radius: 200 ft

*Taken from "Atomic Energy Facts," a United States Atomic Energy Commission publication, available from the Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C.

†Changed from diphenyl.

‡Reduced from 300 psi planned with diphenyl.

Table VII-2 OMRE CORE CHARACTERISTICS[31]

Critical mass at 250°F: 20 fuel elements or 16.4 kg U^{235}
Complete loading: 31 fuel elements or 25.5 kg U^{235}

Reactivity:

Temperature, °F	Reactivity ($\Delta k/k$), %
250	7.87
450–480	8.76
700	7.82

Shutdown reactivity at 250°F:

All six control rod pairs inserted	−8.20
Four outer control-rod pairs inserted Two inner control-rod pairs fully withdrawn	−2.62

Excess reactivity requirements at 700°F:

Equilibrium xenon and samarium	3.88
Fuel burn-up	3.94

rise of 240 to 480°F, an increase of 1 per cent reactivity occurs; any hazard resulting from the positive coefficient could be circumvented by limiting reactor operation to temperatures above 480°F.

The temperature coefficient is, of course, a function of the specific reactor design, and a positive coefficient is not a necessary characteristic of an organic-moderated reactor.

Radioactive Products and Radiation Level

A very significant advantage of the use of an organic coolant is the low induced radioactivity in the pure material. Limited operation of the OMRE, as reported at the SRE-OMRE Forum,[30] indicated that the radioactivity induced in the coolant (modified Santowax O-M) was due to impurities and consisted mainly of the following short-lived gamma emitters.

Isotope	Half life	Concentrations, ppm
Na^{24}	15 hr	0.10
Mn^{56}	2.58 hr	0.03
Cl^{38}	37.3 min	2.00
Cu^{64} Cu^{66}	12.8 hr 5.1 min	0.20

Typical radiation levels, operating at 10,000 kw, obtained around the OMRE facility[30] indicated that the activity problem is small, as indicated by the following: In the organic-coolant reprocessing room, 80 mr/hr, max.; at the coolant exit pipe, 320 mr/hr; above the reactor tank, 1000 mr/hr, max.; in the control room, 0.4 mr/hr.

If no large build-up in activity by crud deposition occurs after long operation, this low contamination should result in large weight savings (relative to the pressurized-water reactor) by requiring little, if any, shielding of the primary loop under normal conditions. However, even in the case of low activity, an evaluation must be made of the hazard produced by a fuel-element failure resulting in fission-product contamination of the coolant. Because of the compatibility of the coolant with the fuel materials this may not be a problem especially in the case of oxide full elements. Besides the savings in weight, the low activity if realized after long-term operation will allow easy access for maintenance.

Reactor Components

All piping and hyraulic components with the exception of the fuel-element clad and the pressure vessel of the OMRE are of carbon steel and are commercial types. This is a direct result of the use of the organic moderator and is a substantial saving in capital cost relative to that of reactors requiring the use of stainless steel throughout. The pressure vessel is made of low alloy steel and retains a pressure of only 200 psi for coolant temperatures of 700°F. Thus the vessel is also significantly cheaper than those of high pressure water reactors, which might be designed for pressures as high as 2,000 psi and employ high alloy steel construction.

It would appear, also, that the corrosion and wear problems will be low and that high reliability and long life of the reactor component may be expected. This is, of course, possibly subject to modification by the long-term effects of the irradiated coolant on these materials and components.

Corrosion

Qualitative data have been presented by Atomics International on the in-pile corrosion by organic polyphenyls of mild steel, aluminum, stainless steel, magnesium, and zirconium specimens.[31,32] These data show that magnesium

and zirconium undergo some corrosion, by oxide formation in the case of the magnesium and by hydride formation in the case of zirconium. Stainless steel and aluminum exhibited insignificant weight change and a slight discoloration of the surface. The mild steel also did not gain a significant amount of weight, but the surface of the specimen was blackened considerably by the hydrocarbon.

Since only qualitative information is available at this time, it is not possible to predict quantitatively the corrosion behavior of the various materials or to determine conclusively whether any problem will develop from the possible transport of corrosion products through the primary coolant loop. However, the tests that have been made are very encouraging.

Safety

The compatibility of the organic liquids with the other reactor materials eliminates the possibility of a hazardous internal chemical reaction. The hot organic liquid does not represent a large amount of stored mechanical energy at the operating temperatures visualized. Therefore, the only rapid energy release that would result from a hypothetical accident would be that of the nuclear reaction. The probability and magnitude of a possible nuclear-energy release will vary according to the particular reactor design, but the magnitude in any case would probably not be large. The question of whether secondary containment would be required could be answered only in relation to a specific reactor design and location. The problems of ensuring integrity of such an envelope should, however, be much less than in the case of a water reactor.

The organics are pyrophoric and burn with dense black clouds of smoke, which make fire fighting difficult around the reactor. Provisions would probably have to be made in the reactor area for an automatic fire-extinguisher system. However, the dust cloud ignition temperature of the organic is quite high (1200°F).

Organic Material

Much of what has been discussed has indicated a decided advantage in the use of an organic moderator or coolant, provided the material is stable under irradiation. As was pointed out previously, although many organic materials have been studied,[1,2] the major emphasis is now being placed on the polyphenyl hydrocarbons.

The OMRE was originally to be cooled and moderated by diphenyl. Later in-pile loop studies indicated greater stability for the terphenyl group, and a decision was made to use a commercial organic of the terphenyl group, Santowax-R (~12 wt.% *o*-terphenyl, ~60 wt.% *m*-terphenyl, and ~28 wt.% *p*-terphenyl). Santowax-R, however, had a melting point between 230 and 300°F, whereas the preheat system for OMRE was designed for the lower melting point of the diphenyl. Because of this circumstance, a lower melting organic, a modified Santowax O-M, (~65 wt.% *o*-terphenyl, ~33 wt.% *m*-terphenyl, and ~3 wt.% *p*-terphenyl) was used. The properties of this material and the pure polyphenyls are shown in Table VII-3, VII-4, and VII-5.

Table VII-3 PROPERTIES OF OMRE COOLANT[31]

Composition (modified Santowax O-M):			
Dipheny, %	16		
o-terphenyl, %	46.1		
m-terphenyl, %	31.8		
p-terphenyl, %	6.1		
Properties:	Temperature		
	500°F	700°F	800°F
Specific gravity	0.91	0.81	0.77
Specific heat, Btu/(lb)(°F)	0.52	0.59	0.60
Viscosity, centipoise	0.46	0.23	0.17
Thermal conductivity, Btu/hr-ft°F	0.069	0.063	0.061
Vapor pressure, psia	5.3	42.9	94.5
Melting point, °F	Solid at 70°F; slurry turning to liquid at 200°F		

It is useful to compare the characteristics of organic coolants with those of other coolants. Such a comparison (Table VII-6) shows that the organic coolant has much poorer heat-transfer properties than either sodium or water.

Although the data in the Table VII-6 are for diphenyl, they do not differ greatly from those for the Santowax materials. The poorer heat-transfer properties with respect to water may be offset by other characteristics of the organics. For example, the compatibility with a wider range of materials may make possible the use of extended surfaces on the fuel elements. The higher temperature capabilities of the organics can, of course, be used to allow a higher temperature drop between the fuel surfaces and

Table VII-4 POLYPHENYL PROPERTIES*

Polyphenyl	M.p., °F	B.p., °F	Vapor pressure, psia 500°F	700°F	900°F	Microscopic cross sec., cm^{-1}	Fission age to thermal at 430°F, cm^2
Diphenyl	157	492	16	110	400	0.00895	72.0
o-terphenyl	135	630	3	27	132	0.00895	69.3
m-terphenyl	189	689	1	16	77	0.0091	67.0
p-terphenyl	416	709	1	12	60	0.00921	65.2

*From unclassified AEC report TID-7007(Pt. 1), pp. 115 and 130, and *Nucleonics*, 14(12): 40-43 (1956), and *Nuclear Engineering*, 2: 524 (1957).

Table VII-5 ORGANIC COOLANT FIRE AND EXPLOSIVE HAZARDS*

	Diphenyl	Santowax-R
Flash point temp., °F	223	360
Flame point temp., °F	255	375
Dust cloud ignition temp., °F	1200	1200
Minimum explosive concentration, oz/cu ft	0.035	0.035

*From unclassified AEC report TID-7007(Pt. 1), pp. 241-242.

Table VII-6 COMPARISON OF SOME COOLANTS*

(Conditions: Flow parallel to an array of fuel rods; Equivalent diameter of flow channel = 0.5 in.; Flow velocity = 20 ft/sec; Mean coolant temperature = 500°F)

	Sodium	Water	Biphenyl Normal viscosity	Biphenyl Twice normal viscosity†
Pressure, psia	14.7	2,000	250	250
Heat-transfer coeff., Btu/(hr)(sq ft)(°F)	15,200‡	6,030‡	1,420	1,080
Heat removal from 0.75-in.-diam. uranium rod relative to that for sodium	1.00	0.93	0.67	0.61
Pumping power relative to pumping power for sodium	1.00‡	0.76‡	0.87	1.0
Pumping power per unit of heat removal relative to that for sodium	1.00	0.82	1.30	1.64
Coolant temp. rise relative to that for sodium	1.00	0.29	0.36	0.33

* From unclassified AEC report TID-7007(Pt. 1), p. 126.
† Twice normal viscosity is shown because of the increase in viscosity that occurs with irradiation.
‡ Taken from *Nucleonics*, 11(6), 27-34 (June 1953).

the coolant, and the increase of efficiency over that obtainable with the lower temperature water-cooled reactor would allow the use of a larger fraction of the power for coolant circulation.

Since the basic advantage of the organic lies in its use at high temperatures, the decomposition of the material at high temperatures under irradiation and the effect on its properties are probably the most important factors in determining the feasibility of its use.

In-pile tests of the polyphenyls have indicated that the magnitude of the decomposition for a given organic is proportional to the total energy absorbed by the organic. The proportionality factors connecting the rate of decomposition, G_p and G_g, with the energy absorption, vary with the percentage of HBR (high-boiling resi-

dues or polymers having higher boiling temperatures than the original material) in the organic.

$$G_p = \frac{\text{molecules of HBR formed}}{\text{100 ev of energy absorbed}}$$

$$G_g = \frac{\text{molecules of gas formed}}{\text{100 ev of energy absorbed}}$$

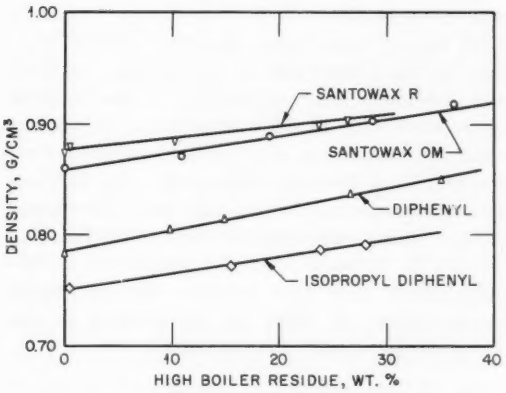

Fig. 7—Density of irradiated polyphenyls at 600°F. (Data taken from reference 31.)

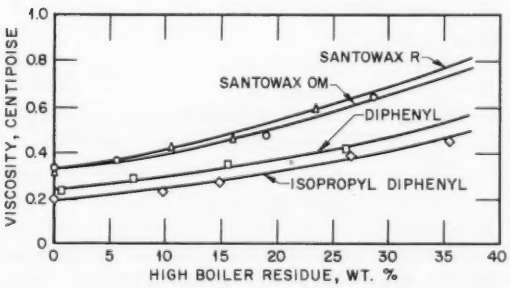

Fig. 8—Viscosity of irradiated coolants at 600°F. (Data taken from reference 31.)

As would be expected, thermal neutrons have little effect on the decomposition; the decomposition is due primarily to gamma rays and epithermal neutrons. Experiments appear to indicate that the G values are nearly the same for neutron- and gamma-energy deposition.

As a result of the breakdown in the organic moderator and the formation of high boiler residues (HBR), the properties of the organics undergo changes. The changes[31] in various properties as a function of HBR content are illustrated in Figs. 7 to 13. More detailed data on the effect of temperature and HBR on the properties of the individual organics are available in reference 31.

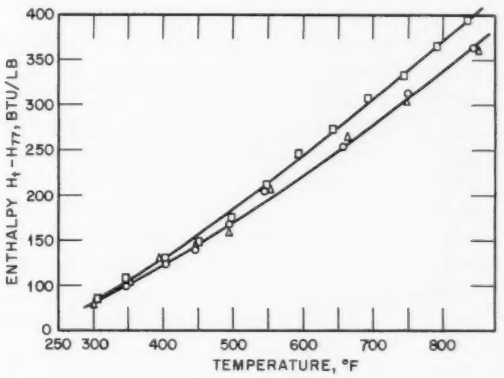

Fig. 9—Enthalpy of polyphenyls. □, 89 per cent *o*-terphenyl, 11 per cent *m*-terphenyl. ○, irradiated Santowax O-M, 32%, HBR. △, OMRE coolant. (Data taken from reference 31.)

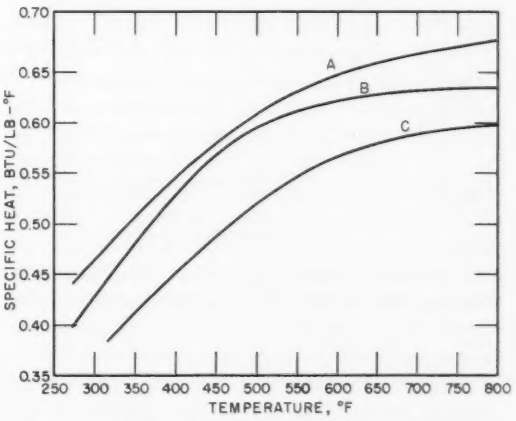

Fig. 10—Specific heat of polyphenyls. Curve A, diphenyl (Forrest et al.). Curve B, 89 per cent *o*-terphenyl and 11 per cent *m*-terphenyl. Curve C, irradiated *o*-terphenyl, Santowax O-M, 32% HBR. OMRE coolant. (Data taken from reference 31.)

Heat transfer and fouling (deposition of coke-like deposits on heat-transfer surfaces) tests were made by Atomics International[31] on diphenyl, Santowax-R and Santowax O-M, circulated at bulk temperatures from 500 to 750°F, at velocities varying from 5 to 25 ft/sec, and at heat fluxes up to 291,000 Btu/(sq ft)(hr),

and an attempt was made to correlate the heat-transfer data by means of the Sieder-Tate equation

$$\nu = 0.027\ (R_e)^{0.8}\ (P_r)^{1/3}\ (\mu/\mu_s)^{0.14}$$

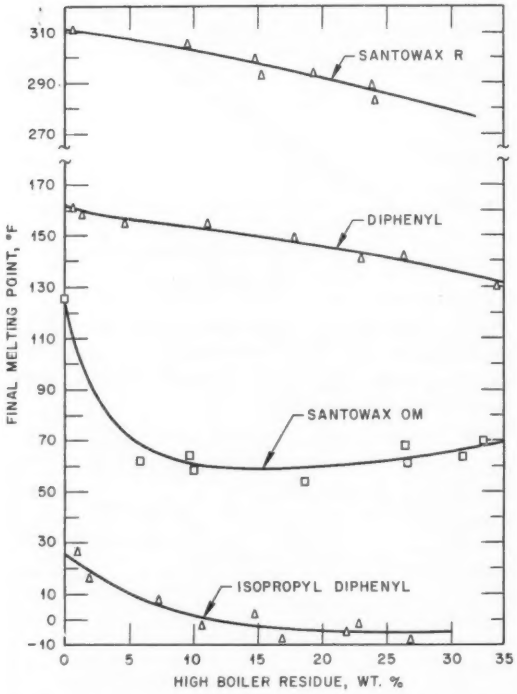

Fig. 11—Melting point (liquidus) of irradiation polyphenyls. (Data taken from reference 31.)

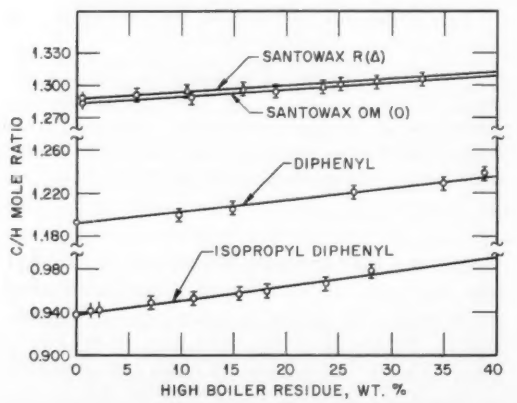

Fig. 12—(Data taken from reference 31.)

Since this equation did not fit the data adequately, a better correlation was established using the following equation

$$\nu = 0.015\ (R_e)^{0.86}\ (P_r)^{0.27} \pm 7.9\%$$

No fouling of the heat-transfer surfaces occurred during tests by Atomics International at heat-transfer surface temperatures of 850 to 1000°F, bulk coolant temperatures of ~760°F, and fluid velocities of 5, 10, and 15 ft/sec for periods up to 100 hr.[31]

As can be seen from the effect of HBR content on the properties of the coolant for best heat transfer, it is preferable to operate with a low percentage of HBR polymer in the coolant while maintaining a high coolant temperature. However, the decomposition rate, G_p, for low HBR concentration is very high and it increases rapidly with temperature above 800°F. Figure 14 is indicative of the decrease in decomposition rate that occurs with increased concentration of HBR in the coolant in the temperature range 600 to 700°F.

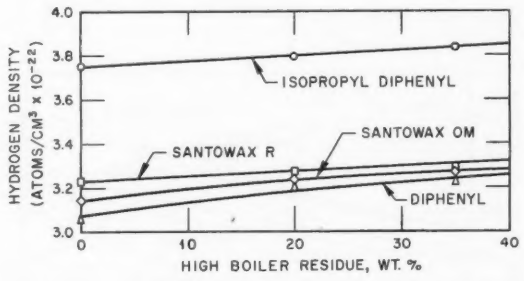

Fig. 13—Hydrogen density of irradiated polyphenyls at 600°F. (Data taken from reference 31.)

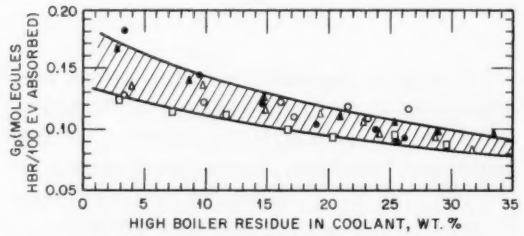

Fig. 14—Thermal and radiolytic decomposition rate of polyphenyls. △, diphenyl (first of three cycles). ▲, diphenyl. ○, Santowax R. □, mixture of Santowax O-M. ●, isopropyl diphenyl. (Data taken from reference 31.)

Above 800°F, it has been found that the decomposition rate increases substantially and thus sets a limit on the practical operating temperature of the coolant. An example of the effect of coolant temperature, above and below 800°F, on the decomposition rate is shown in Table VII-7. The decomposition of the organic coolant may be divided into two components: (1) degradation caused by radiation and (2) degradation caused by purely thermal effects. Thermal decomposition becomes significant only at bulk temperatures greater than 800°F.

Table VII-7 STATITRON (1-MEV ELECTRONS) IRRADIATED *m*-TERPHENYL*

Polymer, %	750°F, G_p (molecules/100 ev)	840°F, G_p (molecules/100 ev)
15	0.082	0.22
20	0.074	0.192
25	0.062	0.178
30	0.055	0.155
35	0.048	0.12
40	0.045	0.078
45	0.035	0.048

* Taken from unclassified AEC report TID-7007(Pt. 1), p. 201.

Operation of the OMRE over several months with HBR concentrations up to 30 per cent has shown no evidence of fouling at the heat-transfer surfaces.

Preliminary results of tests with the OMRE have yielded a decomposition rate of approximately 75.5 lb/Mwd or a $G_p \approx 0.1$ at a coolant temperature of 600°F. In calculating G_p, an average HBR molecular weight of 460 was used. Because of the particular design of this core, which involves a large ratio of organic to fuel and structural material, the average energy absorption in the organic was high (8.4 per cent of the total fission energy).[30,31] The gaseous decomposition products, consisting mainly of hydrogen, methane, ethane, and normal propane, occur at a rate $G_g \approx 0.02$. These decomposition rates appear to substantiate the previous in-pile loop tests, and such data may probably be applied to full-scale reactor designs.

The substantial decrease in the decomposition rate with HBR content (Fig. 14) must be evaluated against the decrease in heat-transfer properties and the possibility of long-term fouling effects. It has been estimated by Atomics International that approximately 30 wt.% HBR is tolerable. At this concentration the decomposition rate $G_p \approx 0.1$.

Reprocessing Cost

The major effect of the dissociation of the organic is the increase in operating cost because of the requirement that moderator material be reprocessed and replaced. The reprocessing system in the OMRE uses a fairly simple single-stage low-pressure distallation unit. The capital cost for the reprocessing will, therefore, be quite low, making the replacement value of the organic the major cost. Typical costs of commercial grade organic coolants have been reported by Atomics International as follows:

	Dollars/lb
Diphenyl	0.15
Santowax-R	0.17
Santowax O-M	0.75
Isopropyl diphenyl	1.25

In the case of the OMRE using a Santowax O-M type organic and assuming a $G_p = 0.1$ at the operating concentration, the replacement material required will be $\sim 3.15 \times 10^{-3}$ lb/kwhr (thermal). It is expected that a commercial grade Santowax-R can be used at a cost of ~17 cents per pound. Using Santowax-R and assuming a replacement rate the same as Santowax O-M, the OMRE organic replacement costs would be ~0.54 mills/kwhr (thermal) or approximately 1.8 mills/kwhr (electrical). To compensate for such an increase in operating cost by decreased capital cost would require a decrease of ~$112/kw (electrical) of installed capacity (based on an average capital charge of 12 per cent per year and a plant factor of 0.85; or 1.6 mills/kwhr per $100 of capital per kilowatt of power capacity).

Atomics International has estimated that for a large (100 Mw electrical) organic-moderated reactor plant the replacement cost of the moderator would be approximately 1.0 mills/kwhr (electrical). Assuming such a cost involves the use of 17 cents per pound Santowax-R and a $G_p = 0.1$ can be obtained at the operating temperature of the reactor, the decomposition rate for a 30 per cent efficient plant must be equal to $\sim 1.77 \times 10^{-3}$ lb/kwhr (thermal) or an energy absorption of the moderator of only

4.5 per cent of the total energy. This lower energy absorption may be obtainable in the larger reactors designed with a higher ratio of fuel and structural material to organic. However, the poor heat-transfer properties of the organic will probably require larger coolant passages than would be required with a water-cooled reactor. The capital cost saving equivalent in this case to offset this operating cost would be ~$63/kw (electrical) or 6.3 million dollars for the 100-Mw reactor. Such a cost could conceivably be saved by the use of the cheaper construction materials and more standard low-pressure equipment.

To further evaluate the worth of the replacement organic, a calculation can be made giving the ratio of the weight of organic per unit of power output when used as a fuel oil to the weight of replacement organic per unit of reactor power. In the case of a reactor having an efficiency of 30 per cent and an organic replacement cost of 1 mill/kwhr (electrical), and on the assumption that the organic cost is 17 cents per pound, the replacement organic per unit of thermal energy of the reactor is equal to 0.0017 lb/kwhr (thermal). Assuming the organic can be burned to produce 18,500 Btu/lb (same as Bunker C oil), the weight per unit of energy is equal to 0.185 lb/kwhr thermal. The ratio of these two which can be looked at as the thermal power gain of the replacement organic is 105.

Although the projected replacement cost for the organic coolant is high enough to represent a substantial contribution to the cost of power, it is still sufficiently low to support the expectation that net economic advantages may be achievable provided no unexpected difficulty arises in cleaning up the organic or in maintaining a clean system.

In applications where weight reduction is important, such as nuclear ship propulsion, the organic-cooled and -moderated reactor may enjoy substantial advantages. These may be sufficient to outweigh, by a comfortable margin, the cost of organic replacement.

In large central station applications it may be desirable to employ a separate moderator and to use the organic as a coolant only. Such an arrangement would improve the neutron economy if a low-absorption moderator such as D_2O or graphite were used. It should also reduce the rate of radiation damage since the rate is proportional to the energy absorption in the organic and a dilution by other materials (particularly by neutron moderating materials), would reduce the fractional absorption in the organic. The same type of effect occurs if the organic is used as coolant and moderator in a lattice which has a high fuel-to-moderator ratio, but in this case the main effect is the absorption of a major fraction of the gamma energy by the fuel, and much of the neutron energy is still deposited in the organic.

Although the experiments to date have gone a long way toward establishing the characteristics of organic liquids as reactor materials, a number of questions remain which affect the design of an economically attractive power reactor. These questions involve the performance of the organic over extended periods of time in an operating reactor. The results of the OMRE will therefore continue to be of great interest.

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A recent paper[1] discusses the neutron flux distortions caused by the distribution of steam in large boiling-water reactors and some of the methods by which these distortions can be controlled.

The rate of steam production in the steam-producing section of a boiling reactor will vary over the volume of the reactor in proportion to the power density. Since reactors of this type have negative steam coefficients of reactivity, the local power density will in turn tend to decrease in those regions where the steam content is large. Consequently, the spatial distribution of power in such a reactor will be modified from that which would exist if no steam were present. In particular, the radial power distribution will tend to flatten, and, since the steam content increases with height in the core, the power distribution will tend to shift toward the bottom of the core. These effects will be small for small reactors because the distortion of the power distribution from the fundamental mode results in higher neutron leakage and the reactivity of the small reactors is quite sensitive to changes in leakage. In the larger reactors, like the Dresden reactor, the distortion may be quite large. Weil[1] has discussed the effect in such a reactor. He points out that the effect is modified in a dual-cycle reactor because the water entering the core is rather highly subcooled and must absorb a fairly large amount of heat before it can reach the boiling point. The exit steam fraction in such a reactor will usually be higher than that in a simple direct-cycle reactor. Furthermore, in a large reactor of this type the power density may be so low in the region near the circumferential boundary that no steam at all is produced. This effect has been reduced in the Dresden reactor by orificing the water flow to the outer fuel elements. The orificing results in a flatter radial distribution of steam production and a flatter average axial power distribution but a somewhat less flat radial power distribution.

Other possible methods mentioned for controlling the power distribution are control-rod scheduling and spatial variation of lattice characteristics and enrichment.

The use of control-rod programming to minimize the effects of the axial power distortion by steam in the Dresden reactor is probably contemplated because the control rods move into the reactor core from the bottom.[2] Thus the absorbing effect of the rods, when they are partially inserted, will tend to make the reactivity of the lower portion of the core more nearly equal to that of the upper portion and will therefore decrease the asymmetry of the power distribution.

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GAS-COOLED GRAPHITE-MODERATED POWER REACTORS

The Atomic Energy Commission has issued a report[1] which summarizes the results and conclusions of work undertaken pursuant to Public Law 85-162, Section 101, on graphite-moderated gas-cooled power reactors. The work was an integrated effort, involving Kaiser Engineers as prime contractor, with ACF Industries, Inc., as subcontractor, and with supporting research and parallel design studies by Oak Ridge National Laboratory and Hanford Atomic Products Operation of the General Electric Company. The more important conclusions of the report are summarized below:

a. There is an adequate basis of technology in the United States for the initiation of construction projects for either natural uranium fueled or partially enriched uranium fueled, civilian, gas-cooled power reactors which are at least as advanced in design concept as any reactors of this type now being built. However, concurrent with any construction project, it would be necessary to undertake extensive research and development on fuel elements, whether metallic uranium with Magnox cladding for the natural version or non-metallic uranium with stainless steel or beryllium cladding for the partially enriched version.

b. The potential of gas-cooled, graphite-moderated reactors designed for use of partially enriched fuel is significantly better than for those designed for natural uranium fuel. This advantage, coupled with the more direct applicability of U. S. technology to the higher temperature designs based on partially enriched fuel, established the preferred direction of any U. S. program which might be carried out in this field.

c. Emphasis on designs using partially enriched fuel would place this technology in the same class with other U. S. reactor designs using enriched fuel from the standpoint of fuel supply problems in other countries. However, reactors using heavy water as moderator are being developed as part of the overall U. S. Civilian Power Reactor Program and while even these reactors are expected to make minimum cost power with some fuel enrichment, they are believed to be more nearly optimum with natural uranium fuel.

d. Nuclear power plants using gas-cooled, graphite-moderated reactors with partially enriched fuel appear to have a potential for making economically competitive nuclear power in the United States. Cost estimates as a function of time have many uncertainties and actual experience will depend upon where, by whom, and under what arrangements the plants are built.

e. Additional research and development, including actual construction and operating experience of a prototype reactor, would be necessary to provide an improved assessment of the potential of this reactor concept.

f. Helium has special attraction, among gases, as a power reactor coolant in graphite moderated reactors designed to operate at high temperatures. However, the use of helium may present a problem of availability (particularly in many other countries), in addition to technical problems of equipment design.

The engineering and design studies are AEC reports, available as a series of six reports[2–7] in twelve volumes from the Office of Technical Services, Washington, D. C. The studies involved both prototype and full-scale plants using natural and enriched uranium. The characteristics of the design studies by Kaiser-ACF and ORNL are summarized in the following paragraphs. In subsequent paragraphs three major sources of technical problems associated with these designs are discussed along with their solutions.

Design and Cost Comparisons

The major characteristics of the designs are compared in Table IX-1, along with three British designs for reference; the estimated plant construction costs and power costs for the three large-scale United States designs are compared in Table IX-2. This comparison illustrates some of the important differences between the natural-uranium and partially enriched reactors of the graphite-moderated gas-cooled type. For a given power capacity the natural-uranium reactors are considerably larger and have higher capital costs; however, they make possible low fuel costs even at rather modest fuel exposures. On the other hand the enriched reactors have lower capital costs but achieve low fuel cost only if the fuel exposure is quite long.

Table IX-1 SUMMARY OF U.S. AND U.K. GAS-COOLED POWER REACTOR DESIGN CHARACTERISTICS[1]

	Kaiser-ACF				ORNL	United Kingdom		
	Natural uranium		Partially enriched			Natural uranium		
	Prototype	Optimum	Prototype	Optimum	Optimum*	Calder Hall	Bradwell	Hinkley Point
Rated thermal output, Mw	180	700	125	600	700	182	530	980.
Rated net electric output, Mw	55	220	44	215	225	35	150	250.
Net thermal efficiency, %	30.6	31.4	35.3	35.8	32.1	19.0	28.3	25.5
Specific power, Mw/metric ton	1.85	2.55	6.9	7.9	5.1	1.4	2.21	2.65
Enrichment, at.%	0.71	0.71	3.0	2.5	2.0	0.71	0.71	0.71
Uranium loading, metric tons	97.5	274	18.0	77	137	130	240	370.
U^{235} inventory, kg	692	1900	540	1925	2740	925	1700	2630
Fuel material	U metal†	U metal†	UO_2	UO_2	UO_2	U metal	U metal	U metal
Cladding material	Magnox A-12	Magnox A-12	S.S. 304	S.S. 304	S.S. 304	Magnox C	Magnox A-12	Magnox A-12
Average fuel exposure, Mwd/metric ton	1500–3000‡	1500–3000‡	10,000	10,000	7350	3000	3000	3000
Refueling scheme, during:	Operation	Operation	Operation	Operation	Shutdown	Shutdown	Operation	Operation
Core diameter by height, ft	35 × 25	50 × 29	12.2 × 18.5	20 × 26	20 × 30	31 × 21	40 × 26	49 × 25
No. of fuel channels	1532	3536	344	980	1597	1695	2620	4500
Coolant gas	CO_2	CO_2	CO_2	CO_2	He	CO_2	CO_2	CO_2
Coolant pressure, psia	280	275	387	370	300	110	165	
Coolant outlet temperature, °F	700–800†	750–800†	1000	1000	1000	637	734	707.
Coolant inlet temperature, °F	409	413	463	473	450	285	356	
No. of coolant loops	2	8	2	6	4	2	6	
Pressure vessel I.D. height, ft	36 × 65	70	18 × 26	27 × 57	50	37 × 70	67	80.
High-pressure steam pressure, psia	1450	1450	2400	2400	950	210	765	650
High-pressure steam temperature, °F	750	750	950	950	950	500	704	685
Low-pressure steam pressure, psia	300	300	750	750	§	63	209	195.
Low-pressure steam temperature, °F	750	750	950	950	§	340	704	670

* Partially enriched uranium.

† Single-region uranium metal–Magnox rod cores limited to 1500 Mwd/ton at 800 or 750°F at 3000 Mwd/ton. Alternate two-region core also proposed; natural uranium in UO_2 with stainless-steel cladding for inner region and uranium metal with Magnox A-12 cladding for outer region to permit 3000 Mwd/ton at 800°F.

‡ High pressure; low pressure.

§ Single pressure cycle.

It is fortunate that the uranium oxide fuel element, whose use imposes the requirement of enriched fuel in this reactor type, appears capable of long life from the standpoint of radiation-damage resistance.

Table IX-2 RECONCILED* GAS-COOLED POWER REACTOR COST ESTIMATES[1]

	Kaiser Engineers-ACF		ORNL
	Optimum, natural uranium	Optimum, enriched uranium	Optimum, enriched uranium
Plant characteristics:			
Net electric power, Mw	220	215	225
Fuel exposure, Mwd/metric ton	3,000	10,000	7,400
Plant construction cost, millions of dollars:			
Total direct costs	72	54	48
Ratio of total to direct costs†	1.75	1.75	1.75
Total design and construction costs	126	95	84
Other capital costs‡	15	11	10
Total capital costs	141	106	94
Unit plant cost, $/kw, net electrical	640	490	420
Power generation cost, mills/kwhr:			
Fixed charges, 14% per year, 80% plant factor	12.8	9.8	8.4
Operation and maintenance	1.2	1.1	1.1
Fuel	2.2	2.5	2.6
Total generation cost	16.2	13.4	12.1

*Reconciled to common basis by provision of equivalent auxiliary facilities and equivalent additive cost factors.

†Total costs include 15% indirect cost, 15% escalation, 20% contingencies, and 10% engineering and design.

‡Includes interest during construction and working capital.

Fuel Materials

It is perhaps not surprising that the preferred reactor design is one utilizing enriched uranium. The fundamental reason for this result is the fact that there is presently no developed fuel-element jacket which allows outlet gas temperatures above about 800°F and which has sufficiently low neutron-absorption characteristics to permit operation with natural-uranium fuel in a graphite-moderated reactor. Apparently, the British are anticipating development of beryllium jackets to perform this function. If economic beryllium jackets are made, the British will have accomplished a straightforward progression to a reactor having both improved thermal performance and good neutron economy by way of an initial step involving a reactor of good neutron economy but limited thermal performance. The higher temperature capabilities of beryllium cannot, however, be fully exploited with metallic uranium fuels of existing types. If oxides or other low-density fuels must be used, then operation with fuel of natural enrichment is probably impracticable even if low-absorption fuel jackets are available.

Certainly the United States should have strong reservations as to the advisability of entering the gas-cooled reactor field by way of a graphite reactor employing fuel elements of the present British type, whose limitations are well known and whose potentialities have perhaps already been exploited nearly to their limit. The alternative fuel element, if the choice is limited to types of more or less proved feasibility, is that element which the report recommends—a steel-jacketed UO_2 element. This type should give good thermal performance at the expense of rather poor neutron economy. If employed as an interim measure, its use may stimulate the development of a better element and lay the foundation for the use of such better elements as they are developed.

Meanwhile, the fuel situation for the proposed reactor looks very much like that for current water-cooled reactors. It is true that the gas-cooled fuel elements are somewhat larger in size and therefore cheaper to fabricate on a unit weight basis and that tolerances and environmental conditions might be sufficiently different to give some further improvements in fabrication costs; nevertheless, the fuel cycle has the already well-known characteristic of partially enriched cycles—the requirement of very high fuel exposure to achieve reasonably low fuel cost in the face of the added expense of chemical processing and the higher net burn-up cost. Uncertainties as to the long-range costs of chemical processing and fuel enrichment still exist, as does the reactor manufacturer's problem of providing his customer some assurance that the long fuel lifetime necessary for acceptable fuel cost can be accomplished. If the possibility of developing a fuel element of excellent neutron economy does

not materialize, then it appears that, in going from the water-cooled to the gas-cooled graphite reactor, only an improvement in steam conditions may be accomplished. Such an improvement in itself is certainly effective in reducing power costs through improved thermal efficiency and through lower specific cost for the steam turbine machinery, but it does not obviously ensure a net economic gain in the face of the poorer heat-transfer and heat-transport properties of gases and the consequent larger sizes of both the reactor and its circulating system.

For both the prototype and the optimum natural-uranium reactor studies of Kaiser-ACF, the preferred fuel assembly is a 2-ft-long cored cylinder of natural metallic uranium encased in a helically finned magnesium alloy can that is held at the center of a cylindrical graphite sleeve by graphite spiders at each end. This graphite sleeve provides structural support for the individual fuel elements within the coolant channel. The assemblies are stacked vertically in fuel channels of the reactor core, nine for the prototype and eleven for the optimum. Longitudinally finned cans were considered, but

Table IX-3 KAISER-ACF AND ORNL PREFERRED FUEL ASSEMBLIES[2-6]

	Kaiser-ACF		ORNL
	Natural uranium*	Enriched uranium*	Enriched uranium†
Fuel slug or pellet:			
Material	U metal	UO_2	UO_2
U^{235} enrichment, at.%	0.71	3.0, 2.5‡	2.0
Geometry	Cored rod	Cored, sintered pellet	Cored, sintered pellet
Outside diameter, in.	1.160	0.750	0.750
Inside diameter, in.	0.375	0.320	0.320
Length, in.	24.00	0.70	0.50
No. per element	1	35	75
Fuel jacket or capsule:			
Geometry	Helically finned tube	Finless tube	Capsule
Material	Magnox A-12	S.S. 304	S.S. 304
Outside diameter, in.	1.27	0.80	0.80
Wall thickness, in.	0.05	0.020	0.020
Length, in.	24.4§	27.5§	38.5
Fin-tip diameter, in.	2.25		
Fuel elements per cluster	1	7	7
Cluster container or hanger:			
Geometry	Open-end sleeve with support spacers (spiders)		Top support hanger, bottom spacer
Material			
Sleeve or hanger	Graphite	Graphite	S.S.
Support spacers	Graphite	S.S.	S.S.
Sleeve diameters, in.	2.50 I.D. × 4.25 O.D.	3.00 I.D. × 3.75 O.D.	
Fuel assembly:			
Length	26.4	29.0	40.0
No. per channel	9,11‡	6,9‡	6
Channel diameter, in.	4.50	4.00	Varying¶

* Prototype and full scale.
† Full scale.
‡ Prototype, full scale.
§ Excluding end caps.
¶ 345 at 3.45 in.; 400 at 3.25 in.; 852 at 3.05 in.

analysis showed this configuration to be inferior to the helically finned concept. Other designs that received consideration were fuel assemblies with: (1) a cluster of three longitudinally finned fuel elements instead of a single helically finned fuel element, (2) a finned graphite support structure surrounding an unfinned can, instead of a helically or longitudinally finned magnesium alloy can, and (3) uranium dioxide slugs with unfinned stainless-steel jackets; fuel assemblies with uranium dioxide were used only in core arrangements in which metallic uranium fuel assemblies were also used. The predicted life of the preferred fuel assembly is 3000 Mwd/metric ton of uranium, provided that the prescribed loading pattern is followed and that the surface temperature does not exceed 850°F.

The fuel slugs are made from wrought-uranium rods, cut to length and bored. The magnesium alloy cans are either extruded by ordinary means or impact-extruded. The can is hydrostatically pressed around the fuel slug to form a mechanical bond between the uranium and the magnesium alloy. The graphite sleeves are extruded, graphitized, and machined to the required dimensions, whereas the graphite spiders are molded or machined.

Design characteristics for the Kaiser-ACF and ORNL fuel assemblies are given in Table IX-3.

The Kaiser-ACF enriched-uranium fuel assembly is a cluster of seven identical columns of stacked UO_2 pellets in individual stainless-steel cans or tubes supported within a graphite sleeve by top and bottom stainless-steel spiders (Table IX-3). The centers of the six outer fuel rods of the cluster are equally spaced on a 2-in. circle. A magnesium oxide spacer pellet, 0.19 in. long and 0.70 in. in diameter, is placed at each end of each column of fuel pellets for thermal insulation of the end caps of the jacket.

The fuel and insulating pellets are jacketed by type 304 stainless-steel cans, having welded top and bottom end caps of the same material, with ends shaped and slotted to fit into the spiders. The fuel elements and spiders are surrounded by a graphite sleeve which may require silicon carbide or other suitable coating if CO_2 is used as coolant. Internal shoulders machined at each end of the sleeve receive the top and bottom spiders.

The ORNL fuel-element assembly consists of one top hanger, seven fuel capsules, and one bottom spacer, as described in Table IX-3. The top hanger is supported at two points 180 deg apart in circumferential recesses in the graphite channel. Loading will consist in sliding the hanger into vertical grooves in the graphite channel and rotating the assembly to engage the hanger in the circumferential recesses. The fuel capsule is type 304 stainless steel, filled with UO_2 slugs and having hemispherical end caps with suitable attachments for hanging and positioning the capsule. Components required for fabricating a fuel capsule are one tube, one top cap, one bottom cap, two MgO end spacers, and 75 UO_2 fuel slugs.

Coolant

The first consideration for a coolant gas must be its chemical compatibility with the materials of the reactor at the temperatures of operation. The most straightforward approach to this requirement would be the use of one of the chemically inert gases. Of these, only helium and argon are available at reasonably low cost (see Table IX-4). Argon is rather unattractive because it gives poor heat-removal performance and it activates strongly under neutron bombardment. Helium gives good performance and in many respects is an attractive coolant. Its availability is restricted geographically, and its cost, although relatively low (Table IX-4), is yet high enough that sizable losses from the coolant system could not be tolerated economically. Aside from these considerations the disadvantages of helium stem from its low molecular weight. The most important of these

Table IX-4 COST AND AVAILABILITY OF GASES[†]

Gas	Cost, $/cu ft*	Availability
He	0.015†	U. S. (Italy and USSR)
A	0.074‡	0.93 vol.% air§
Ne	42.50¶	0.0018 vol.% air§
CO_2	0.003¶	World wide
CO	0.24**	World wide
H_2	0.10¶	World wide
N_2	0.08¶	78.06 vol.% air§

* At standard temperature and pressure.
† F.O.B. Amarillo, Tex.
‡ F.O.B. Richland, Wash.
§ Obtained by fractional distillation of liquid air.
¶ F.O.B. Kennewick, Wash.
** F.O.B. New Jersey.

Table IX-5 PLANT COMPONENT DESCRIPTION

	Optimum Kaiser-ACF		ORNL
	Natural	Enriched	Enriched
Net electrical output, Mw	700	600	700
Primary coolant piping:			
Number of parallel circuits	8	6	4
Pipe inside diameter, ft	4	3	5
Circuit pressure drop, psi	17	40	7
Coolant blowers:*			
Number	8	6	4
Total electrical pumping power, Mw	25	28	18
Steam generators:			
Type	Dual pressure reheat		Single pass
Number	8	6	4
Approximate over-all dimensions, ft	10 diam. × 90	10 diam. × 90	20 diam. × 60

*Motor driven, axial flow.

is the high rate of leakage of helium through small openings and the consequent difficulty of achieving acceptably low leakage rates in a pressurized system.

Gases other than the chemically inert ones must be considered in relation to the temperatures at which they are to be used, to the other materials in the system, and to possible modification of the normal chemical behavior by radiation effects. Carbon dioxide is the only gas whose performance has been established by use in a reactor. It is compatible with the Magnox alloys to about 450°C, with aluminum up to temperatures at which the aluminum becomes unusable because of loss of strength, and with stainless steel up to 800 or 900°C. It begins to react appreciably with graphite, forming carbon monoxide, at temperatures in the neighborhood of 500°C.

As shown in Table IX-1, the coolants employed in the Kaiser-ACF and ORNL designs are carbon dioxide and helium, respectively. The final decision as to which coolant should be used was not made in these studies. It is generally agreed that carbon dioxide is desirable provided the problem of compatibility with graphite at high temperature can be solved economically.

Components

As a result of the design studies, it was reaffirmed that physical plant construction, including fabrication of large pressure vessels, should not be a technical limitation to plant capacity, particularly for the partially enriched-uranium designs; capacities in excess of 200 Mw (electrical) are feasible. Although gas leakage rates and some equipment costs are important unknowns, no serious development problems in connection with providing steam generators, blowers, valves, and other plant equipment are indicated.[1]

In continuance of the natural-uranium–enriched-uranium design comparison, Table IX-5 describes some of the major plant components and supplements Table IX-1.

References

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2. Gas-cooled Power Reactor, Preliminary Design, 55,000 Kw Prototype Natural Uranium Nuclear Power Plant, IDO-2021(Rev.), March 1958. (Unclassified AEC report.)
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6. The ORNL Gas-cooled Reactor, ORNL-2500, March 1958. (Unclassified AEC report.)
7. A Parametric Study of the Gas-cooled Reactor Concept, HW-54727, March 1958. (Unclassified AEC report.)

From the standpoint of neutron economy, heavy water is the best of all practical moderators. Its absorption is low enough that most D_2O reactors can tolerate reasonably large amounts of additional absorption by structural and fuel jacketing materials and still exhibit rather high neutron economy. Since the moderating power ($\xi\Sigma_S$) is not high (compared to that of H_2O), relatively large ratios of moderator-to-fuel volume are typically used in natural-uranium or slightly enriched D_2O reactors. Consequently, the volume of D_2O required for moderation is usually much larger than the volume which would be required in the reactor to accommodate sufficient coolant for the fuel elements. A number of coolants could therefore conceivably be used with D_2O moderation without departing drastically from the high neutron economy which is characteristic of D_2O. D_2O-moderated reactors using a number of different coolants are presently receiving attention in the United States and elsewhere.

The major advantage anticipated for the D_2O reactor results from the high neutron economy. For example, the D_2O-moderated reactor appears to be the type that has the highest probability of achieving the optimum economic operation with natural-uranium fuel. Although there is not yet a D_2O-moderated power reactor in existence, the reactor type has been used extensively for other purposes such as research and test reactors and plutonium-production reactors. A rather extensive background of experience therefore exists in the use and handling of D_2O. Additional problems enter of course when the high-temperature operation necessary for power production is attempted. In the following paragraphs the work that is being done on D_2O-moderated power reactors of various types is described briefly.

D_2O-cooled Reactors

The Canadian work is perhaps the most advanced on this reactor type. A nuclear power demonstration reactor (NPD-2) of approximately 20 electrical megawatts capacity has been designed and is expected to be completed in Canada by 1961. A recent Canadian study[1] has considered the design of a large power reactor (200 Mw net electrical) of this type, which could be built within eight years. The reactor is of the pressure-tube type, with the tubes running horizontally through the moderator tank. The moderator, D_2O, is kept cool by a separate cooling circuit. The fuel elements are of natural UO_2, consisting of cylindrical UO_2 pellets jacketed with Zircaloy-2. The elements, which are about 1 ft long, are assembled into clusters or fuel subassemblies. Clusters of from 19 to 61 elements each are under consideration. The reactor is to be reloaded under power by pushing new fuel subassemblies into the fuel (pressure) tubes and removing spent subassemblies from the opposite ends of the tubes. This push-through type of reloading is done in opposite directions in adjacent fuel tubes. The average composition of the reactor therefore remains nearly constant over the reactor length after the equilibrium fuel cycle has been attained. With this type of fuel reloading a uranium burn-up of 7400 Mwd/ton is expected from the natural UO_2 elements. The general characteristics of the reactor are summarized in Table X-1. The smaller power demonstration reactor (NPD-2) will contain many of the features of this large reactor and may be considered a prototype plant leading to its development.

In the United States, studies and development on natural-uranium-fueled heavy-water-moderated power reactors have been undertaken by E. I. du Pont de Nemours & Company for the Atomic Energy Commission. This work has been under way for about a year and is described in quarterly progress reports.[2–5] The last two of these[4,5] contain summaries of much of the work.

A number of reactor designs have been investigated by Du Pont, including D_2O- and gas-cooled types in both the pressure-vessel and the pressure-tube versions. Selection of the preferred design is based on the requirement that it be capable of being built by mid-1962. The

design judged to require the least research and development has therefore been designated the preferred one. It is a pressure-vessel type, in which the moderator operates at roughly the same temperature as the coolant. The design power is 100 Mw electricity (net). The reactor vessel is about 13 ft in internal diameter and

In its passage through the reactor, the heavy water serves first as moderator and then as coolant for the fuel. The moderator temperature averages about 400°F, the coolant inlet temperature is 403°F, and the outlet temperature is 450°F. Other characteristics of the reactor are given in Table X-1.

Table X-1 SUMMARY OF LARGE NATURAL-URANIUM D_2O REACTOR DESIGNS

	NPD (AECL)[1]	Du Pont (USAEC)[4,5]
Reactor type:		
Fuel	Oxide	Metal
Pressure resisting unit	Tube	Vessel
Coolant and moderator	D_2O	D_2O
Over-all characteristics:		
Reactor thermal power, Mw	794	435
Gross electrical power, Mw	208	106
Net electrical power, Mw	200	100
Net plant efficiency, %	25.2	23.0
Fuel inventory, short tons	77 (oxide)	29.2 (metal)
D_2O inventory, short tons	186	131.3
Fuel life, Mwd/ton uranium	7400	3250
Specific power (fuel), kw/kg	12.8	16.4
Core geometry:		
Lattice pitch, in.	12.0	6.5
Core radius	9.05	5.23
Core length, ft	16.65	15.00
Reflector thickness (radial), in.	12.0 (H_2O)	12.0 (D_2O)
Reflector thickness (axial), in.	12.0 (H_2O)	12.0 (D_2O)
Number of channels	252	340
Fuel assembly:		
Housing I.D., in.	4.47 (pressure tube)	2.90
Housing wall thickness, in.	0.024 (pressure tube)	0.030
Housing material	Zr	Zr
Fuel elements per assembly	31	1
Fuel element clad O.D., in.	0.63	2.06
Fuel clad thickness, in.	0.02	0.03
Fuel O.D., in.	0.587	2.06
Fuel I.D., in.	0.138	1.49
Heat transfer:		
Coolant pressure, psia	1000	547
Moderator pressure, psia	14.7	547
Coolant average temperature in reactor, °F	480	430
Moderator temperature, °F	176	405
Coolant velocity, ft/sec	24.6 (max.)	17.8 (inside) 12.5 (outside)
Maximum fuel temperature, °F	4000	975
Maximum clad temperature, °F	540	565 (inside) 572 (outside)

34 ft high and has walls 3 in. thick and heads of thickness about 7 in. The design pressure of the vessel is approximately 680 psig, and the operating pressure is about 550 psia. The material is low-alloy carbon steel lined with stainless steel.

The reactor contains 340 fuel assemblies, each of which consists of a single Zircaloy-clad tube of natural uranium which is alloyed with 2 wt.% zirconium. The assemblies are surrounded by Zircaloy housing tubes to direct the coolant flow. The tubular fuel elements, which are being

developed by Nuclear Metals, Inc., are produced by coextrusion. The use of these single large extrusions as fuel elements is expected to play an important role in achieving low fuel-fabrication costs.

Although the comparison of the reactor characteristics in Table X-1 is useful for illustrating some of the design trends of pressure-vessel and pressure-tube type reactors, an absolute comparison of performance characteristics is misleading because the Du Pont design is based on the assumption of construction at an early date, whereas the Canadian design is based on the assumption that construction should be completed within about eight years. In consideration of the very substantial improvements that might be made through further research and development, the Du Pont group has proposed that a small reactor (61 Mw thermal) be constructed before the large one as a means of testing fuel elements and as a means of developing those improvements that are foreseen for the reactor type. The proposed reactor is basically similar in design to the large 100-Mw (electrical) reactor which was the basis of the design study.

Another small (70 Mw thermal) heavy-water-moderated and -cooled reactor is to be built at Hanford[6] for use in the plutonium recycle program. This reactor is a test reactor, having specific experimental functions to perform in the recycle program, but it will at the same time furnish worthwhile information on the reactor type.

The reactor is of the pressure-tube type, and the moderator is contained in an unpressurized aluminum calandria tank. The cylindrical-shaped calandria is approximately 10 ft in height and 7 ft in diameter. Passing vertically through the calandria are 85 fuel channels, 18 shim-control channels, and 13 flux monitor channels. The 85 fuel channels are arranged in an equilateral triangular lattice. The thermal insulation between the hot fuel channels (the pressure tubes) and the moderator is provided by the gas space between the pressure tubes and the calandria tubes.

The fuel elements are installed inside the pressure tubes and are cooled by upward flow of the D_2O coolant. The pressure tubes are individually piped to a single, circular water header at each face of the reactor. For experimental purposes the tubes can be individually monitored or piped to separate cooling systems. The pressure tubes are made of Zircaloy-2 with a wall thickness of 0.154 in. They are 3.25 in. in inside diameter and are separated from the calandria tubes by a clearance of $\frac{1}{4}$ in. The pressure tubes are tapered to a smaller size at the lower end to simplify fitting design and tube replacement in the event of damage or fuel-element failure.

Although the reactor heat is not used to generate power, the temperatures used are typical of those for a power reactor. The coolant inlet temperature is 479°F, and the outlet temperature is 530°F. The heat is removed from the primary coolant in a boiler type heat exchanger and is then dissipated to the river.

Primary control of reactivity is obtained by adjusting the level of the heavy-water moderator within the calandria. Absorber type shim controls are provided which consist of 36 neutron absorbing rods, each 3 ft long, spaced in pairs through the reactor core. Each rod is moved individually by a loop of miniature stainless-steel stud chain driven through a reduction gear train by a high-speed electric motor. The rods operate in the cool moderator of the reactor. Rapid emergency shutdown is provided by rapid drainage of the moderator from the calandria.

The fuel elements for the reactor will be equal in length to the reactor core (7 to 9 ft). They will have thin-walled Zircaloy-2 jackets filled with either sintered uranium oxide or plutonium-aluminum alloy. They will take the form of either clusters of 19 small rods or assemblies of two coaxial tubes plus a central rod. After initial operation with fuel elements of these types, new types containing different core materials may be used for test and evaluation purposes.

The development, construction, and operation of a tube type D_2O-cooled D_2O-moderated power reactor have recently been undertaken by the Carolinas-Virginia Nuclear Power Associates, Inc.[7] This reactor is to operate at a thermal power of 60.5 Mw and will employ, in addition, oil-fired superheat. Steam produced by the plant will be capable of generating approximately 19 Mw (gross) of electrical power.

Sodium Cooled Reactor (SDR)

The first status report[8] on the sodium-cooled D_2O-moderated reactor project, whose objective is to install a 10-Mw electrical SDR power plant

in Anchorage, Alaska, was presented at the Nuclear Engineering and Science Conference in Chicago on Mar. 17 to 21, 1958.

Fundamentally, the combination of D_2O moderation with sodium cooling would appear to have considerable interest from the point of view of reactor performance. The undesirable feature of this combination is the chemical incompatibility of sodium and water. The central engineering problem in the design of such a reactor is to guarantee that substantial quantities of sodium and water do not come together under any conditions of operation or malfunction of the reactor and to accomplish this end without adding so much neutron absorber to the reactor that the good nuclear properties of D_2O are nullified.

All sodium reactors that have been designed to date transfer heat from sodium to water in their steam systems, and, to some extent, the problem of keeping sodium and water separated is therefore not a new one. When the sodium and deuterium exist together in the reactor vessel itself, however, the situation is complicated by three new considerations.

1. A restriction is placed upon both the types and quantities of materials which may be used as separators between the two fluids by considerations of neutron economy.

2. It is no longer possible to isolate the water from the radioactive sodium by employing an intermediate sodium system, as is usually done in sodium-cooled power reactors.

3. A coupling is introduced between nuclear accidents and chemical accidents. For example, the melting of fuel either by a nuclear excursion or because of a deficiency of coolant could conceivably contribute to the failure of the barrier between sodium and D_2O.

The approach to the problem of guaranteeing separation of sodium and D_2O is exemplified by the description of the reactor design. The D_2O moderator is contained in a calandria, 8.4 ft high and 10 ft in diameter, which consists of an upright cylindrical aluminum tank penetrated by vertical aluminum tubes fastened to the tank top and bottom. There are 120 tubes, 4 in. in diameter and on 8-in. centers. A steel fuel-coolant tube containing a cluster of steel-jacketed fuel elements passes through each aluminum tube. The sodium coolant flows upward through the fuel tubes, entering and leaving each tube through individual smaller tubes, or pigtails, which provide flexibility to accommodate thermal expansion. The pigtails are collected into manifolds which in turn feed into header pipes. Into the gas-filled annular space between each fuel coolant tube and its surrounding calandria tube, one or more concentric barrier tubes may be inserted to provide additional mechanical barriers for the separation of sodium and D_2O.

The nominal reactor power is 40 Mw thermal. Fuel loading is 7.7 tons, and the fuel is expected to have an enrichment of 2 per cent or less of U^{235}.

The sodium inlet temperature is 750°F, and the outlet temperature is 950°F. The average D_2O temperature is 150°F.

The experimental program aimed toward solving the problem of sodium-D_2O separation was described. It encompasses investigations of materials to be used for sodium barriers between the fuel tubes and the calandria tubes, methods of incorporating such barriers, methods of designing the sodium system for maximum integrity, and mock-up tests. An encouraging result of the barrier material test is that aluminum shows relatively good performance for this application. When exposed to a jet of high temperature sodium, a 0.01-in.-thick sample of type 1100 aluminum sheet showed slight distortion after 15 min exposure at 950°F. A 0.060-in.-thick sheet of the same material showed no attack after 16 hr at 950°F and required about $2\frac{1}{2}$ hr to erode through at a temperature of 1100°F. Another part of the investigation concerned the relative effects of oxygen and inert atmospheres when sodium and water are reacting. Simple tests, involving the dropping of small pellets of sodium into water, showed that explosions occurred between oxygen and the hydrogen liberated by the sodium-water reaction when the atmosphere above the water surface was air. No explosion occurred when argon was substituted for the air.

Organic-cooled Reactor

The organic-cooled D_2O-moderated reactor has recently received some attention.[9]

The use of an organic liquid as reactor coolant only, rather than as coolant and moderator, has the advantage of minimizing the radiation-induced damage to the organic because the damage will be very nearly proportional to the fraction of radiation energy absorbed by the organic in competition with the other materials present

in the reactor (the structure, fuel, and moderator). It would in fact appear that the heavy-water-moderated reactor might well be one of the most attractive applications for the organic coolant. The low vapor pressure of the organic would appear to be particularly attractive for a tube type reactor, the type considered in the reference. Although the heat-transfer properties of organic liquids suitable for reactor coolants are known to be considerably poorer than those of water, the authors point out that in many current reactor designs the heat conductivity in the fuel rather than the heat removal by the coolant is the limitation on specific power, and they state that they have made designs of organic-cooled reactors that have specific powers in the range 15 to 20 thermal kilowatts per kilogram of fuel, a range comparable to that which characterizes present water-cooled designs. Furthermore, in a D_2O reactor the use of organic coolant might well reduce the D_2O inventory relative to that of a D_2O-cooled design, even if the specific power with respect to fuel were considerably poorer than in the D_2O-cooled design. The authors contemplate using natural-uranium metal as fuel. The reactivity gain from the use of metal rather than oxide would undoubtedly more than compensate for the absorption of the organic relative to D_2O coolant. The authors favor the aluminum alloy M-257, which contains approximately 10 per cent Al_2O_3 as the material for the fuel tubes. The fuel elements would probably be finned for improved heat transfer, and aluminum alloy or magnesium would be considered for the fuel jacket material.

As more results become available from the Organic Moderated Reactor Experiment (see (pp. 42-51), they will undoubtedly be examined in relation to the D_2O-moderated reactor and perhaps in relation to other reactor types employing separate moderators.

One of the problems that was not discussed in the reference was that of achieving a negative coolant void coefficient of reactivity. This can undoubtedly be done by designing the reactor for sufficiently high resonance absorption, but it is not apparent without analysis that the desired coefficient could be achieved at the same time as operation on natural uranium.

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Reference 1 contains a series of papers which were presented at the Industrial Preview of the Engineering Test Reactor (ETR) at Idaho Falls, Idaho, on Oct. 2 and 3, 1957. The papers contain information of interest to potential users of the facility, ranging from the design and operating characteristics of the reactor and the experimental facilities to the operating procedures and the general procedures for obtaining test time in the reactor.

Until the completion of the ETR,[2,3] the MTR was the major test reactor in the United States, at least from the standpoints of magnitude of operating power and available neutron flux. The ETR represents a large increase in the amount of high-flux test space available, as well as an increase in the flux level available in the test holes. These increases have necessarily been accomplished at the expense of a large increase in operating power (to 175 Mw) over that of the MTR. Aside from the increase in operating power, the major change in approach to the irradiation problem in ETR has been the incorporation of extensive irradiation facilities in the core of the reactor. This is evidently the effective way of utilizing the neutron flux produced; provision was not made for the installation of large test loops in the MTR core for several reasons, the most important of which was that the reactor was not originally designed primarily as an engineering test reactor.

In relation to the development of power reactors, one of the important considerations is the cost of building and operating test reactors. An estimate of the operating cost for the ETR has not been given, but the operating cost of the MTR is stated to be approximately $600 per hour, including all costs such as direct labor, fuel, materials, maintenance, amortization over 25 years, technical assistance, and overhead expenses for both Phillips Petroleum Co. (the operating contractor) and the Atomic Energy Commission, but excluding operation and maintenance of experimental equipment which is furnished by experimenters as part of the experiment.[1] The construction cost of ETR is itemized in reference 1 and amounts to 17.3 million dollars; this figure includes engineering, development, and inspection.

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